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NewYorkPower Authority

James Knubel
Senior Vice President and
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August 16, 2000
IPN-00-059

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Proposed Improved Technical Specifications
Reply to NRC Request for Additional Information

REFERENCE:

1. NRC letter, "Request for Additional Information Regarding STS Conversion," G. Wunder to J. Knubel, dated July 9, 1999.
2. NYPA letter IPN-98-139, "Proposed Technical Specification Change, Conversion to ITS," J. Knubel to NRC, dated December 15, 1998.
3. NRC letter, "Schedule for Submitting Additional Information on Improved STS," G. Wunder to J. Knubel, dated July 11, 2000.

Dear Sir:

The Authority is providing responses to Requests for Additional Information (Reference 1) regarding Revision 0 of the proposed Improved Technical Specifications for Indian Point 3 (Reference 2). This transmittal addresses the following ITS Sections.

1.0 Use and Application
2.0 Safety Limits
3.0 LCO / SR Applicability
3.1 Reactivity Controls (consists of 8 subsections)
3.2 Power Distribution (consists of 4 subsections)
3.7 Plant Systems (consists of 17 subsections)
3.9 Refueling (consists of 6 subsections)

Attachment I outlines the revision status for each of the ITS sections based on the following change categories.

- Changes required to address NRC RAIs
- Changes required to incorporate new amendments to the IP3 current Technical Specifications
- Changes or corrections proposed by the Authority

A001

Attachment I also identifies whether Revision 1 of the proposed ITS conversion package is needed based on the scope of the above changes. Attachment II is the Authority's reply to each of the RAIs for the ITS sections addressed by this transmittal. Attachment III contains Revision 1 pages for the proposed ITS conversion packages, if needed.

The Authority plans to submit Revision 1 changes for the remaining ITS sections by September 25, 2000. Proposed replies to RAIs for all ITS sections have already been discussed with the NRC staff and inputs to the safety evaluation table were provided prior to July 31, 2000 to support an amendment date of December 15, 2000 (Reference 3).

The Authority is making no new commitments in this letter. If you have any questions, please contact Mr. Ken Peters.

Very truly yours,

J. Knubel
Senior Vice President and
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cc: Regional Administrator
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REVISION STATUS FOR PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

ITS NUM	ITS SECTION TITLE	NRC RAIs	New Amendment	NYPA Changes	COMMENT
1.0	Use and Application	8	Yes No impact	None	Deleted definition of PTLR and reclassified DOCs per RAIs. Submittal of Revision 1 proposed ITS not required.
2.0	Safety Limits	0	Yes No impact	Yes	Editorial correction to Part 1; Figure 2.1-1 should be same as CTS not STS. Correct Figure will be included in final ITS. Submittal of Revision 1 proposed ITS not required.
3.0	LCO / SR Applicability	0	None	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.1	REACTIVITY CONTROL	(7)			
3.1.1	Shutdown Margin	0	Yes No impact	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.1.2	Core Reactivity	1	Yes No impact	Yes	Revised ITS package to change Applicability per RAI. Also, NYPA editorial correction noted in Bases. Revision 1 of proposed ITS submitted for review.
3.1.3	Moderator Temperature Coef	0	No	Yes	NYPA revision to Bases page. Markup of STS page B 3.1-20 submitted for review.
3.1.4	Rod Group Alignment Limits	2	Yes see comment	Yes	Changes reflect RAI response and incorporate Amendment 197. NYPA proposed changes consist of plant specific clarifications in the Bases. Revision 1 of proposed ITS submitted for review.
3.1.5	Shutdown Bank Insertion Limits	0	No	No	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.1.6	Control Bank Insertion Limits	0	No	No	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.1.7	Rod Position Indication	3	Yes see comment	Yes	Changes reflect RAI response and incorporate Amendment 197. NYPA change consists of revision to Bases. Revision 1 of proposed ITS submitted for review.
3.1.8	Physics Tests Exceptions	1	Yes No impact	No	Bases change explained in reply to RAI. Submittal of Revision 1 proposed ITS not required.

REVISION STATUS FOR PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

ITS NUM	ITS SECTION TITLE	NRC RAIs	New Amendment	NYPA Changes	COMMENT
3.2	POWER DISTRIBUTION LIMITS	(5)			
3.2.1	Heat Flux Hot Channel Factor	1	Yes No impact	No	Various changes to Spec, Bases, and DOCs to incorporate RAI reply. Revision 1 of proposed ITS submitted for review.
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	0	Yes No impact	No	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.2.3	Axial Flux Difference	1	No	No	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.2.4	Quadrant Power Tilt Ratio	3	No	No	Various changes to Spec and Bases to incorporate RAI reply. Revision 1 of proposed ITS submitted for review.

ITS NUM	ITS SECTION TITLE	NRC RAIs	New Amendment	NYPA Changes	COMMENT
3.7	PLANT SYSTEMS	(30)			
3.7.1	Main Steam Safety Valves	1	Yes No impact	None	Specification and Bases revised to reflect RAI response. Rev 1 of proposed ITS (Parts 1 and 5) submitted for review
3.7.2	MSIVs and MS Check Valves	2	No	None	Bases revised to reflect response to RAI 3.7.2-02. Affected Revision 1 Bases pages submitted for review.
3.7.3	MBFP Discharge and Regulation	1	No	None	Specification and Bases revised to reflect RAI response. Rev 1 of proposed ITS (Parts 1 and 5) submitted for review
3.7.4	Atmospheric Dump Valves	1	No	None	Editorial changes to the Bases per RAI response. Submittal of Revision 1 proposed ITS not required
3.7.5	Auxiliary Feedwater System	4	Yes No impact	None	Specification and Bases revised to reflect RAI response and correction made to CTS markup page. Affected Revision 1 pages submitted for review.
3.7.6	Condensate Storage Tank	3	Yes No impact	Yes	Various changes to Specification, Bases, and DOCs to incorporate RAI reply. NYPA change to Bases for Actions A.1 and A.2. Revision 1 of proposed ITS submitted for review.
3.7.7	City Water	0	Yes No impact	None	Editorial change to Bases (removed reference to FSAR sections 6 and 14) as explained in reply to RAI 3.7.6-3. Submittal of Revision 1 proposed ITS not required.

REVISION STATUS FOR PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

ITS NUM	ITS SECTION TITLE	NRC RAIs	New Amendment	NYPA Changes	COMMENT
3.7.8	Component Cooling Water	2	Yes No impact	None	No changes to Specification or Bases. Added DOC LA.3, L.3 (with No Significant Hazards Evaluation), and JFD X.1 per RAI replys. Affected Revision 1 pages submitted for review.
3.7.9	Service Water System	4	Yes No impact	None	Various changes to Specification, Bases, DOCs and JFD to incorporate RAI reply. Revision 1 of proposed ITS submitted for review.
3.7.10	Ultimate Heat Sink	1	Yes No impact	Yes	Specification, Bases and DOCs changed in response to NRC comment from 3/8/00 conference call. Additional change to Bases proposed by NYPA. Affected Revision 1 pages submitted for review.
3.7.11	Control Room Vent. System	4	No	Yes	Editorial corrections to Bases per RAI response and additional Bases changes proposed by NYPA. Revision 1 Bases pages provided for review.
3.7.12	Control Room Air Cond. Sys	3	No	None	Bases revised to incorporate response to RAI. Affected Revision 1 pages submitted for review.
3.7.13	FSB Emergency Vent. Sys	1	Yes No impact	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.7.14	Spent Fuel Pit Water Level	0	No	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.7.15	Spent Fuel Pit Boron Conc	1	Yes No impact	Yes	Correction to Bases per RAI response. NYPA proposes change in SR frequency. Affected Revision 1 pages submitted for review.
3.7.16	Spent Fuel Assembly Storage	1	Yes No impact	Yes	Correction to Bases Figure per RAI response. NYPA change consists of correction to Part 5 insert. Affected page submitted for review.
3.7.17	Secondary Specific Activity	1	Yes No impact	No	No changes to Specification or Bases. New L-DOC and associated NSHE provided for review per RAI response.

REVISION STATUS FOR PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

ITS NUM	ITS SECTION TITLE	NRC RAIs	New Amendment	NYPA Changes	COMMENT
3.9	REFUELING OPERATIONS	(8)			
3.9.1	Boron Concentration	3	Yes No impact	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.9.2	Nuclear Instrumentation	2	Yes No impact	None	No changes to Specification or Bases. DOCs revised to incorporate reply to RAIs. Revision 1 DOCs submitted for review.
3.9.3	Containment Penetrations	3	Yes No impact	Yes	Revision to Specification and DOC to incorporate reply to RAIs. NYPA changes consist of editorial corrections to Part 5 inserts and addition of an M-DOC. Affected Revision 1 pages submitted for review.
3.9.4	RHR & coolant circ - Hi Lvl	0	Yes No impact	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.9.5	RHR & coolant circ - Lo lvl	0	Yes No impact	Yes	NYPA change consists of an editorial correction to ITS clean page. Affected page submitted for review.
3.9.6	Refueling Cavity Water Level	0	Yes No impact	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.

ATTACHMENT II TO IPN-00-059

NEW YORK POWER AUTHORITY REPLIES TO

NRC REQUESTS FOR ADDITIONAL INFORMATION

REGARDING REVISION 0

OF PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

The following ITS Sections are addressed:

ITS	TITLE	RAIs
1.0	Use and Application	8
2.0	Safety Limits	none
3.0	LCO / SR Applicability	none
3.1	Reactivity Controls	7
3.2	Power Distribution Limits	5
3.7	Plant Systems	30
3.9	Refueling	8

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **1.0 USE AND APPLICATION**

NRC RAI No: **1.0--01**

RAI STATEMENT:

--DOC A.3:

The licensee is proposing to add the following definitions to the ITS that are not part of the CTS:
--Actions
--Trip Actuating Device Operational Test (TADOT)
--Mode
--La
--Leakage
--Axial Flux Difference (AFD)
--Physics Tests
--Pressure and Temperature Limits Report (PTLR)
--Slave Relay Test
--Staggered Test Basis
--Master Relay Test

Adding new definition to the ITS is not considered an administrative change.

Comment: Provide a revised DOC with proper categorization to justify these changes.

NYPA RESPONSE:

NYPA agrees with reclassifying ITS 1.0, Discussion of Change (DOC) A.3, as a More Restrictive Change (DOC M.4). The basis for classification as a More Restrictive Change is that adding a definition more explicitly defines a requirement; therefore, adding definitions is a more restrictive change.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **1.0 USE AND APPLICATION**

NRC RAI No: **1.0--02**

RAI STATEMENT:

--CTS 1.2.2, Hot Shutdown Condition
--DOC A.4, A.5

The categorization of these DOCs are incorrect. Because adding new definitions and languages to the ITS is not an administrative change. While the DOCs provide adequate technical justification, the proper category designation is still necessary.

Comment: Provide revised DOCs with proper categorization to justify these changes.

NYPA RESPONSE:

NYPA agrees with reclassifying ITS 1.0, Discussion of Change (DOC) A.4 and A.5, as More Restrictive Changes (DOC M.6 and M.7, respectively). The basis for classification as a More Restrictive change is that the clarifications to the CTS definition more explicitly defines a requirement.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **1.0 USE AND APPLICATION**

NRC RAI No: **1.0--03**

RAI STATEMENT:

--CTS 1.9.2, Instrument Channel Functional Test
--CTS 1.9.3, Instrument Channel Calibration
--CTS 1.9.4, Logic Channel Functional Test
--DOC A.12. A.13, A.14, A.15

The categorization of these DOCs are incorrect. Because adding new definitions and languages to the ITS is not an administrative change. While the DOCs provide adequate technical justification, the proper category designation is still necessary.

Comment: Provide revised DOCs with proper categorization to justify these changes. (Same comment as for RAI question 1.0-2)

NYPA RESPONSE:

NYPA agrees with reclassifying ITS 1.0, DOCs A.12, A.13, A.14 and A.15 as DOCs L.3, L.4, M.7 and M.8 to address the following issues:

-the ITS will allow the use of an actual signal in addition to a simulated signal to initiate a Functional Test;

-the ITS explicitly states that only required alarms, displays and trip functions must be included in a channel Operational test (COT) and that the test must include adjustments as necessary to ensure that required functions are within the required range and accuracy;

-the ITS Definition of Channel Calibration clarifies that calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in place qualitative assessment of sensor behavior and normal calibrations of the remaining adjustable devices in the channel; and

-The ITS Definition of Channel Calibration clarifies that whenever a sensing element is replaced, the next required Channel Calibration must include an in place cross calibration that compares the other sensing elements with the recently installed sensing element.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **1.0 USE AND APPLICATION**

NRC RAI No: **1.0--04**

RAI STATEMENT:

--DOC A.16

These DOCs are to explain the addition of new definition to the ITS. Adding new definition to the ITS is not considered an administrative change. This is an issue of a similar nature with RAI question 1.0-1.

Comment: Provide a revised DOC with proper categorization to justify these changes.

NYPA RESPONSE:

NYPA agrees with reclassifying ITS 1.0, DOC A.16 as a More Restrictive Change (DOC M.9). The basis for classification as a More Restrictive Change is that adding ITS Section 1.2 - Logical Connectors; Section 1.3 - Completion Times; and, Section 1.4 - Frequency more explicitly defines existing requirements; therefore, adding these ITS sections is a more restrictive change.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **1.0 USE AND APPLICATION**

NRC RAI No: **1.0--05**

RAI STATEMENT:

--CTS 1.7, Protection Instrumentation and Logic
--DOC A.12

The entire definition in CTS 1.7 is proposed to be deleted, and DOC A.12 is designated to provide the justification for this deletion. However, DOC A.12 does not address CTS 1.7, but it addresses for CTS 1.9.2 definition of Instrument Channel Functional Test.

Comment: Clarify this confusion and, if necessary, provide a DOC of proper categorization to support the proposed changes made to CTS 1.7.

NYPA RESPONSE:

CTS 1.7, Protection Instrumentation and Logic, is mis-labeled. CTS 1.7 should have been labeled DOC A.2. DOC A.2, which already lists CTS 1.7, deletes CTS definitions because the current Technical Specifications that use these definitions are not retained in the ITS (and deletion or re-location of the specification is addressed elsewhere in the submittal), or the equivalent ITS Specification will not use the defined term.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **1.0 USE AND APPLICATION**

NRC RAI No: **1.0--06**

RAI STATEMENT:

--CTS 1.17, Core Operating Limits Report

The CTS states, "...reload cycle in accordance with Specification 6.9.1.6...," and "6.9.1.6" is proposed to be deleted and replaced with "5.6.5." It appears that no DOC is available to address this change.

Comment: Provide a DOC for this change.

NYPA RESPONSE:

CTS 1.17, Core Operating Limits Report, reference to CTS "6.9.1.6" to ITS "5.6.5" is a numbering change and has been annotated as DOC A.1

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **1.0 USE AND APPLICATION**

NRC RAI No: **1.0--07**

RAI STATEMENT:

---CTS 1.18, Shutdown Margin
--DOC A.18, A.19, A.20

The categorization of these DOCs are incorrect. Because adding new definitions and languages to the ITS is not an administrative change. While the DOCs provide adequate technical justification, the proper category designation is still necessary.

Comment: Provide revised DOCs with proper categorization to justify these changes. (Same comment as for RAI question 1.0-2)

NYPA RESPONSE:

NYPA agrees with reclassifying ITS 1.0, DOCs A.18, A.19 and A.20 as More Restrictive Changes (DOC M.10, M.11 and M.12, respectively). The basis for classification as a More Restrictive change is that the clarifications to the CTS definition more explicitly defines a requirement. with the definition of Shutdown Margin are justified in DOCs M.10, M.11 and M.12.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **1.0 USE AND APPLICATION**

NRC RAI No: **1.0--08**

RAI STATEMENT:

---CTS 3.8, Refueling, Fuel Handling and Storage

The Applicability and the Objective parts of this CTS are proposed to be deleted. However, a DOC is missing from the markup. Also, for the Specification part of this CTS, the markup states, "See CTS master markup," and this statement needs to be clarified with additional information.

Comment: Provide an appropriate DOC for this deletion and clarify the meaning of the statement, "See CTS master markup."

NYPA RESPONSE:

"CTS Master Markup" refers to IP3 ITS submittal Volumes 16 and 17 which consist of a copy of all CTS pages that were annotated to show changes between the CTS and ITS. These annotated CTS pages are arranged in CTS order to facilitate verification that all CTS line items were addressed in the IP3 ITS conversion.

The CTS Master Markup shows that the Applicability and Objective section on CTS page 3.8-1 was deleted by DOC A.2 in the packages for ITS LCOs 3.7.13, 3.9.2, 3.9.3, 3.9.4 and 3.9.5.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.1.2 Core Reactivity**

NRC RAI No: **3.1--01**

RAI STATEMENT:

ITS 3.1.2 Core Reactivity, Applicability
CTS 3.10.10
DOC L.1 and JFD T.2

The accepted CTS interpretation and practice is that core reactivity specification is applicable whenever the reactor is critical. The STS Applicability is in Modes 1 and 2. The ITS Applicability is in Mode 1, based upon "approved" TSTF-141.

Comment: The vendor has determined that TSTF-141 does not apply to Westinghouse plants. Adopt the STS Applicability of "Modes 1 and 2."

NYPA RESPONSE:

IP3 ITS revised to delete allowance provided by TSTF-141. LCO 3.1.2 Applicability revised to be MODES 1 and 2. Bases revised accordingly. DOC L.1 and JD T.2 deleted.

It appears that approval for TSTF-141 was withdrawn for Westinghouse plants after the IP3 submittal was completed.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.1.4 Rod Group Alignment Limits**

NRC RAI No: **3.1--02**

RAI STATEMENT:

ITS 3.1.4 Rod Group Alignment Limits
ITS SR 3.1.4.1
CTS 3.10.5.1
DOC A.14 and JFD CLB.2

The CTS allows a 1 hour temperature soak, for RPI stabilization, after rod movement. The STS does not have this allowance since the STS does not require rod position verification immediately after rod movement, but instead requires rod position verification on a periodic basis (every 12 hours). The ITS retains the 1 hour soak allowance in a note to SR 3.1.4.1.

Comment: The note is not necessary in the ITS, since require rod position verification is not required immediately after rod movement; ITS SR 3.1.4.1, rod position verification, can be accomplished with a 1 hour soak without the need to stipulate its allowance.

NYPA RESPONSE:

SR 3.0.1 states that "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." Without the Note to SR 3.1.4.1, LCO 3.1.4 is not met and LCO 3.1.4 Actions must be initiated if individual rod positions do not indicate within required alignment limits immediately after completion of rod motion. Therefore, IP3 will retain the Note to SR 3.1.4.1 based on CLB.

NYPA revised the Note to SR 3.1.4.1 to state that the SR is not required to be "met" versus not required to be performed consistent with the intent of the Note. The change is needed because use of the verb performed does not provide the required relaxation that the Note is intended to provide (i.e., the SR would still have to be met during the one hour allowance but it would not necessarily have to be performed).

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.1.4 Rod Group Alignment Limits**

NRC RAI No: **3.1--03**

RAI STATEMENT:

ITS 3.1.4 Rod Group Alignment Limits
ITS SR 3.1.4.3
CTS 3.10.8
JFD PA.1

CTS 3.10.8 requires measuring rod drop time from the "loss of stationary gripper coil voltage," while the STS requires measuring from the "beginning of decay of stationary gripper coil voltage." The ITS retains the CTS wording.
Comment: The STS wording seems to be more definitive, in that some may interpret that stationary gripper coil voltage is not lost until it is decayed away (beyond some point). Justify retaining the CTS wording.

NYPA RESPONSE:

NYPA maintained the wording "from loss of stationary gripper coil voltage to dashpot entry" from CTS 3.10.8 because changing to the criteria in NUREG-1431 could require a change to the test method or at least a technical evaluation of the acceptance criteria. The STS wording implies that the voltage profile is monitored at the stationary gripper coil as opposed to simply removing power to the coil stack. Therefore, the current wording is maintained as a plant-specific wording preference.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.1.7 Rod Position Indication**

NRC RAI No: **3.1--04**

RAI STATEMENT:

ITS 3.1.7 Rod Position Indication
CTS 3.10.6
DOC A.3

CTS 3.10.6.1 permits more than one Rod Position Indication per Bank to be inoperable. The STS with the incorporation of approved TSTF-234 permits more than one Rod Position Indication to be inoperable. The ITS does not allow more than one Rod Position Indication per group to be inoperable.

Comment: Review TSTF-234 for applicability to IP3 and consider adopting.

NYPA RESPONSE:

NYPA has reviewed TSTF-234 and concludes that it is applicable to IP3. NYPA intends to incorporate TSTF-234 in the final version of ITS. This involves the addition of a new Condition B for 'More than one IRPI per group inoperable' , along with Required Actions, Completion Times, and Bases as described in the TSTF.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.1.7 Rod Position Indication**

NRC RAI No: **3.1--05**

RAI STATEMENT:

ITS 3.1.7 Rod Position Indication, Required Action A.1
CTS 3.10.6.1
DOC M.2

CTS 3.10.6.1 requires, with an inoperable RPI, that control rod position be checked (with the excore or incore detectors) once per 8 hours. The STS similarly requires that control rod position be checked (with the incore detectors) once per 8 hours.
Comment: The ITS incorporates a beyond scope change by relaxing subsequent rod position verifications to every 24 hours. This change needs to be justified, and a TSTF change submitted.

NYPA RESPONSE:

NYPA revised ITS to adopt a Completion Time of "Once per 8 hours" for Required Action A.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.1.7 Rod Position Indication**

NRC RAI No: **3.1--06**

RAI STATEMENT:

ITS 3.1.7 Rod Position Indication, Required Action B.1
CTS 3.10.6.1
DOC L.1

CTS 3.10.6.1 requires, with an inoperable RPI, that control rod position be checked (with the excore or incore detectors) after rod motion exceeding 24 steps, with an implied completion time of "immediately." The STS requires, with an inoperable RPI, that control rod position be checked (with the incore detectors) after rod motion exceeding 24 steps, within 4 hours.

Comment: The ITS incorporates a beyond scope change by relaxing the rod position verifications to within 8 hours. This change needs to be justified, and a TSTF change submitted.

NYPA RESPONSE:

NYPA revised ITS to adopt a Completion Time of "4 hours" for Required Action B.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.1.8 PHYSICS TESTS Exceptions MODE 2**

NRC RAI No: **3.1--07**

RAI STATEMENT:

ITS B3.1.8, Bases Background Section
Unidentified Deviation

The list and description of the Physics Tests required for reload fuel cycles is not included in the ITS Bases.
Comment: This deviation from the STS Bases is not justified/explained. It seems that at least a list of the recurring tests would be useful.

NYPA RESPONSE:

NYPA revised ITS so that the ITS B3.1.8, Bases Background Section includes the following list:

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:
a. Critical Boron Concentration - Control Rods Withdrawn;
b. Control Rod Worth;
c. Isothermal Temperature Coefficient (ITC); and
d. Neutron Flux Symmetry.

The Critical Boron Concentration - Control Rods Inserted test will not be included because this test is not always performed.

A description of these tests will not be included in the Bases.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.2.1 Heat Flux Hot Channel Factor (FQ(Z))**

NRC RAI No: **3.2--01**

RAI STATEMENT:

ITS 3.2.1 Heat Flux Hot Channel Factor
STS 3.2.1 Required Actions A.3 and B.1
JFD DB.1 and JFD CLB.1

The CTS does not include STS 3.2.1 Required Actions A.3 (to reduce Overpower Delta T trip setpoints) and B.1 (to reduce AFD limits), and therefore are not included in the ITS.

Comment: It is not clear why STS 3.2.1 Required Actions A.3 and B.1 are not be applicable to IP3; discuss.

NYPA RESPONSE:

Part a:
NYPA revised ITS to adopt a Required Action A.3, "Overpower Delta T - High trip setpoints > 1% for each 1% FCQ(Z) exceeds limit." with a Completion Time of "72 hours."

Part b:
IP3 will not incorporate Condition B (FWQ(Z) not within limits) because this requirement and the associated Action is not currently required at IP3 and applies only to plants that use Relaxed Axial Offset Control operation. IP3 uses Constant Axial Offset Control operation.

This conclusion is supported by the discussion in the LCO Section of the NUREG-1431 Bases which states the following:

"For Relaxed Axial Offset Control operation, FQ(Z) is approximated by FCQ(Z) and FWQ(Z). Thus, both FCQ(Z) and FWQ(Z) must meet the preceding limits on FQ(Z)." and

"The expression for FWQ(Z) is: FWQ(Z) = FCQ(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."

NYPA revised the ITS submittal to include the reason given above for not incorporating STS Required Action B.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.2.3 AXIAL FLUX DIFFERENCE (AFD)**

NRC RAI No: **3.2--02**

RAI STATEMENT:

ITS B3.2.3 AFD
B3.2.3 Bases LCO Section, pages B3.2-30 & B3.2-31
JFD PA.1

The ITS includes an improved LCO Bases section, including insert B3.2-30-01.

Comment: Prepare and submit a TSTF change proposal to revise/improve the STS.

NYPA RESPONSE:

NYPA plans to submit a TSTF change request to add improvements to the Bases for LCO 3.2.3.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.2.4 QUADRANT POWER TILT RATIO (QPTR)**

NRC RAI No: **3.2--03**

RAI STATEMENT:

ITS 3.2.4 QPTR
ITS Bases SR 3.4.2.4
CTS 3.10.2.9
DOC LA.2

The CTS details on using incore detectors ("at least two thimbles per quadrant") are moved to the Bases for SR 3.4.2.4.
Comment: In moving this detail to the Bases, the information in the STS regarding quadrant symmetry has been deleted; why?

NYPA RESPONSE:

NYPA revised ITS SR 3.2.4.2 Bases as follows:

"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."

IP3 Bases will not include the specific detector locations because IP3 has procedural controls for determining locations.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.2.4 QUADRANT POWER TILT RATIO (QPTR)**

NRC RAI No: **3.2--04**

RAI STATEMENT:

ITS Condition A Note
JFD PA.1

The ITS adds a note, not in the STS, that "Required Actions A.4, A.5, and A.6 must be completed whenever Condition A is entered."

Comment: Justify this addition, and prepare a TSTF change proposal to revise the STS.

NYPA RESPONSE:

NYPA revised ITS to delete the Note to Condition A that states: "Required Actions A.4, A.5, and A.6 must be completed whenever Condition A is entered."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.2.4 QUADRANT POWER TILT RATIO (QPTR)**

NRC RAI No: **3.2--05**

RAI STATEMENT:

ITS 3.2.4 QPTR
ITS Condition A Required Actions
JFD PA.1

The ITS does not include approved TSTF-241.
Comment: Review TSTF-241 and consider incorporating into ITS 3.2.4.

NYPA RESPONSE:

NYPA revised ITS to incorporate TSTF-241, R4. "Allow time for stabilization after reducing power due to QPTR out of limit."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.1 Main Steam Safety Valves (MSSVs)**

NRC RAI No: **3.7.1--01**

RAI STATEMENT:

--Bases 3.7.1 ACTIONS--

Comment: How do you obtain 42% as stated in Table 3.7.1-1 for two valves inoperable or the 109% for no valves inoperable using the formula on page B 3.7.1-4?

NYPA RESPONSE:

Required adjustments to the neutron flux trip setpoint when one or more main steam safety valves (MSSVs) are inoperable are based on guidance provided in Nuclear Safety Advisory Letter (NSAL) 94-001, Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves, and Information Notice 94-60, Potential Overpressurization of Main Steam System. The equation used to calculate the setpoint adjustment included in the Bases of Required Action A.1 is from these documents. However, CTS and ITS Bases do not include the stipulation that the results of this calculation " must then be adjusted lower for use in Technical Specification 3.7.1.1 to account for instrument and channel uncertainties (typically 9%). CTS Table 3.4-1 and ITS Table 3.7.1-1 include a very conservative adjustment that exceeds 9%. This difference accounts for the difference in the calculated value and the value in ITS Table 3.7.1-1 when one or more Main Steam Safeties are inoperable.

The calculation shown in the Bases for ITS 3.7.1, Required Action A.1, is not used to calculate the neutron flux trip setpoint when there are no inoperable MSSVs.

NYPA revised ITS to incorporate the following improvements:

(1) ITS Table 3.7.1-1 revised to eliminate the statement of requirements when all (5) MSSVs are Operable. This will eliminate any conflict between the nominal setpoint established in LCO 3.7.1 and the allowable value established in LCO 3.3.1.

(2) Bases revised to state that the values in ITS Table 3.7.1-1 are based on the formula from NSAL 94-001 but also includes a very conservative allowance of 9% to account for instrument uncertainty.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.2 Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)**

NRC RAI No: **3.7.2--01**

RAI STATEMENT:

-- ITS 3.7.2

Comment: Addition of Main Steam Check Valves (MSCVs) is Beyond Scope. Further, the 48 hours for the MSCVs needs additional justification. The Bases for Required Action C.1, in part, justifies the 48 hours for MSIVs by stating that the time allows for the certain repairs of the valves that can be done online. Can the same be said for the MSCVs?

NYPA RESPONSE:

NYPA believes that the IP3 design requires that structure and presentation of IP3 ITS LCO 3.7.2 is approved as submitted. Therefore, this change is within scope. Additionally, NYPA wishes to maintain the 48 hour Allowable Out of Service Time (AOT) for both one or more MSCVs and one MSIV with an 8 hour AOT for one MSIV and one or more MSCVs. NYPA believes that DOC L.1 and the ITS 3.7.2 Bases provide adequate justification for this change. The following summarizes the justifications for the AOTs in IP3 ITS 3.7.2:

1. The 48 hour AOT for one or more inoperable MSCVs (Condition A) is acceptable because for the following: MSCVs provide protection for an upstream break only and provide no protection for a down stream break; only the one MSCV associated with the faulted SG is required to function to provide the required protection; and, proper functioning of the 3 MSIVs not associated with the faulted SG eliminates the need for the MSCV to function. Therefore, a 48 hour AOT for one or more MSCVs is more conservative than a 48 hour AOT for one inoperable MSIV and significantly more conservative than the 48 hour AOT when all MSIVs are inoperable as is permitted by 3.4.B (3.7.2 DOC M.1).

2. The 48 hour AOT for one inoperable MSIV (Condition C) is significantly more conservative than the 48 hour AOT when all MSIVs are inoperable as is permitted by 3.4.B (3.7.2 DOC M.1)

3. The 8 hour AOT for one MSIV inoperable and one or more MSCVs inoperable (Condition F) is consistent with the 8 hour AOT in NUREG-1431 for an inoperable stop-check valve.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.2 Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)**

NRC RAI No: **3.7.2--02**

RAI STATEMENT:

-- Bases ITS SR 3.7.2.2

Comment:. The meaning of Bases insert B 3.7-12-01 is unclear. "... that it closes properly with no steam flow as is required to perform its design function"?

Additionally, insert B 3.7-9-01 states that MSCVs are considered operable when inspections and testing required by Inservice testing are completed at the specified frequency. Doesn't performance of the TS SR have to be included?

NYPA RESPONSE:

1. NYPA revised ITS to delete the phrase: "with no steam flow as is required to perform its design function", from the Bases for SR 3.7.2.2. The description of SR 3.7.2.2 now states: "Each MSCV must be inspected to ensure that it closes properly."

2. NYPA revised ITS (LCO Bases) to read as follows: "The MSCVs are considered OPERABLE when inspections and testing required by the Inservice Test Program are completed at the specified FREQUENCY in accordance with SR 3.7.2.2.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.3 Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (MFRVs) and MFRV Low Flow Bypass Valves**

NRC RAI No: **3.7.3--01**

RAI STATEMENT:

--ITS 3.7.3 and Bases 3.7.3

Comment: In the Applicable Safety Analyses discussion on page B 3.7.3-2 it states that the MBFPDVs and the MBFRVs satisfy Criterion 3. However, there is no discussion of the low flow bypass valves.

Additionally, the inclusion of the word "boiler" in the names of the components does not appear to be consistent with the FSAR. (See main feedwater discussion FSAR pages 10.2-27 thru 10.2-29).

NYPA RESPONSE:

1. NYPA revised ITS to include the following description: The MBFPDVs, MFRVs and MFRV Low Flow Bypass Valves satisfy Criterion 3 of 10 CFR 50.36.

2. Terminology used to describe the Main Boiler Feedpump is inconsistent. NYPA selected Main Boiler Feedpump Discharge Valves (MBFPDV) for consistency with P&ID's and plant labeling. The acronyms have been changed for Main Feedwater Regulation Valves (MFRV) and MFRV Low Flow Bypass.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.4 Atmospheric Dump Valves (ADVs)**

NRC RAI No: **3.7.4--01**

RAI STATEMENT:

--Bases 3.7.4

Comment:
The title used for the bypass system is inconsistent. On pages B 3.7.5-1 and 4-4 it is the "Steam Bypass System".

On pages B 3.7.4-3 and 7-1 it is the "Turbine Steam Bypass System".

On page B 3.7.2-2 it is the "turbine, Steam Bypass System (High Pressure Steam Dump)", on page B 3.7.4-1 it is "Steam Bypass System (High Pressure Steam Dump System)", and

on page 3.7.6-1 it is "steam bypass (High Pressure Steam Dump)".

Additionally, on page B 3.7.4-2 item c) "reticulated flow"?

Finally, in the discussion of A.1 on Bases page B 3.7.4-3 it discusses 30 days when the Completion Time in the TS is 7 days.

NYPA RESPONSE:

NYPA has revised the ITS Bases to consistently use the name "Steam Bypass System" as is used in STS. NYPA includes the clarification "(HP Steam Dump)" because IP3 has another system commonly referred to as "Turbine Bypass System" or "Low Pressure Steam Dump." NYPA is including this clarification in the ITS Bases to prevent confusion as to which system is the subject of this ITS section.

Other typos as noted were corrected.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.5 Auxiliary Feedwater (AFW) System**

NRC RAI No: **3.7.5--01**

RAI STATEMENT:

--ITS SR 3.7.5.4

Comment: In Note 2 there is a typo, it should be "steam generator" not "steamgenerator"

NYPA RESPONSE:

NYPA revised ITS to correct this typographical error.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.5 Auxiliary Feedwater (AFW) System**

NRC RAI No: **3.7.5--02**

RAI STATEMENT:

--Bases 3.7.5

Comment: The first paragraph of the BACKGROUND states "... and pump to the steam generator secondary side that connect to the main feedwater (MFW) piping outside containment." Clearly the MFW piping connects to the steam generator inside containment. Was the intent to say that the AFW system connects to the MFW piping outside containment?

NYPA RESPONSE:

NYPA revised ITS to include the following description: "... and pump to the steam generator secondary side via a connection to the main feedwater (MFW) piping at a point outside containment."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.5 Auxiliary Feedwater (AFW) System**

NRC RAI No: **3.7.5--03**

RAI STATEMENT:

-- ITS LCO 3.7.5 Note

Comment: There is no justification in the CTS or in the submittal for insert 3.7-11-01 "capable of supporting the credited steam generator." It may be acceptable but should be pursued generically through the TSTF.

NYPA RESPONSE:

The IP3 Bases Background and LCO Section explain that the IP3 design differs from the standard plant design in that each of the steam generators can be supplied by one of the two motor driven AFW pumps. Therefore, the requirement to have a motor driven AFW pump in Mode 4 makes sense only if it can be aligned to the SG(s) being credited as the backup decay heat removal method. The LCO Note and the Bases have been revised to change "steam generator" to "steam generator(s)".

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.5 Auxiliary Feedwater (AFW) System**

NRC RAI No: **3.7.5--04**

RAI STATEMENT:

-- ITS SR 3.7.5.3

Comment: There is no justification for the apparent deletion of the six month surveillance frequency of CTS 4.8.1.b.

Additionally, ITS Bases insert B3.7-24-01 notes that the steam driven AFW pump discharge valves must be manually opened by the operator. Therefore, it is unclear how ITS SR 3.7.5.3 accounts for the valves as its scope is limited to automatic valves.

NYPA RESPONSE:

CTS 4.8.1.b, a requirement to cycle the auxiliary feedwater pumps discharge valves at intervals not greater than six months, is relocated to the Inservice Testing Program in ITS 3.7.5, DOC LA.2. The CTS markup page has been corrected to show that CTS 4.8.1.b is relocated by LA.2.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.6 Condensate Storage Tank (CST)**

NRC RAI No: **3.7.6--01**

RAI STATEMENT:

-- ITS 3.7.6

Comment: The CTS Bases specifically require checking the CST breather valves as part of system operability (CTS page 3.4-4). Neither DOC A.1 nor A.4 justify why the breather valves need not be included in the TS when they were specifically part of the CTS.

NYPA RESPONSE:

Requirement for breather valves is in the CTS Bases and is not a Technical Specification.

NYPA revised ITS 3.7.6 LCO Bases (the definition of operability) to include the following:

"The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level. "CST venting and pressure relief capability is required for the CST to perform both its normal and emergency function. The venting and pressure relief functions are satisfied by either of the CST breather valves or equivalent venting capacity."

Additionally, ITS 3.7.6 was expanded to justify deletion of the shutdown requirement when one of the two breather valves is not functional. This justification is provided in DOC L.2.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.6 Condensate Storage Tank (CST)**

NRC RAI No: **3.7.6--02**

RAI STATEMENT:

-- ITS 3.7.6 Required Action A.1

Comment: Since there only two AFW water sources "backup water supply" should be changed to "City Water supply".

NYPA RESPONSE:

NYPA revised ITS 3.7.6, Required Action A.1, to read: "Verify by administrative means OPERABILITY of City Water."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.6 Condensate Storage Tank (CST)**

NRC RAI No: **3.7.6--03**

RAI STATEMENT:

-- ITS 3.7.6 Bases

Comment: Why does TS 3.7.7, which is for the backup supply, reference FSAR chapters 6 and 14 while TS 3.7.6 which is for the primary water supply does not?

Similarly why does ITS 3.7.6 meet Criteria 2 and 3 while 3.7.7 which provides the same function only meet Criterion 3?

NYPA RESPONSE:

1. NYPA revised the ITS to delete the references to FSAR chapters 6 and 14 in ITS 3.7.7. The only significant FSAR reference for either Condensate Storage or City Water is FSAR, Section 10. This reference states: The minimum 360,000 gallons of water in the condensate storage tank is adequate for decay heat removal for a period of at least 24 hours. A back-up source of feedwater is available from the city water system.

The statement in Bases for ITS LCO 3.7.7, CW can be used to provide cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the FSAR, Chapters 6 and 14 (Refs. 2 and 3, respectively) is true; however, the information in FSAR, Chapters 6 and 14, regarding city water is not meaningful.

2. TSTF-140 revised NUREG-1431, LCO 3.7.6 Bases to state that in addition to Criterion 3 (mitigation), CST volume also meets Criterion 2 (process variable assumed as an initial condition). NYPA determined that City Water meets Criteria 3 (mitigation) but does not meet Criterion 2 (process variable assumed as an initial condition) because the minimum amount of water in the condensate storage tank is the amount needed to maintain the plant for 24 hours at hot shutdown following a trip from full power. When the condensate storage tank supply is exhausted, city water will be used.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.8 Component Cooling Water (CCW) System**

NRC RAI No: **3.7.8--01**

RAI STATEMENT:

-- ITS SR 3.7.8.1

Comment: There is no justification for why 92 days was chosen for the frequency of the SR when the STS specifies 31 days.

NYPA RESPONSE:

CTS 3.3.E has no explicit requirements for periodic verification of CCW valve positions. IP3 is voluntarily adopting ITS SR 3.7.8.1 (see 3.7.8, DOC M.2) which requires a periodic verification of valve lineup. NYPA proposes to perform the valve lineup at a 92 day Frequency versus the 31 day Frequency specified in NUREG-1431.

The system is flow balanced and some supported systems use CCW during normal plant operation, i.e., spent fuel pit cooling, etc. Monitoring of CCW flow indicators and other supported systems during normal operation could provide indication of a mis-positioned CCW valve. The CCW loops are normally cross connected during normal and emergency operation; while the cooling loads are divided between the two loops so that each loop is capable of supplying the necessary service to support normal operation heat loads or accident heat loads. Any nonessential required service water system pump can be used to support either or both CCW heat exchangers. Therefore, a single mis-positioned valve may have minimal consequence. NYPA believes that performing ITS SR 3.7.8.1 at a 31 day Frequency would be a significant burden without a commensurate contribution to plant safety.

NYPA revised ITS to include a justification as JFD X.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.8 Component Cooling Water (CCW) System**

NRC RAI No: **3.7.8--02**

RAI STATEMENT:

-- CTS 3.3.E.1.b and CTS 3.3.E.2.b

Comment: The submittal indicates that consideration of these CTS requirements was moved to ITS 3.5.2. Although there is a mention of them in the ITS 3.5.2 Bases, the applicable pages from the CTS were not provided in the ITS 3.5 portion of the submittal nor was any discussion provided that justifies moving these two CTS requirements to the ITS Bases.

NYPA RESPONSE:

NYPA revised ITS to include ITS 3.7.8, DOC L.3, that justifies that the requirements of CTS 3.3.E.1.b and CTS 3.3.E.2.b affect only the Operability of the Containment Recirculation Pumps. Therefore, these requirements are covered in the Bases of ITS 3.5.2 and CTS 3.3.E.1.b and CTS 3.3.E.2.b. The corresponding CTS 4.5.B.1 for the associated surveillance is relocated to the Inservice Test Program as stipulated in DOC LA.3.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.9 Service Water System (SWS)**

NRC RAI No: **3.7.9--01**

RAI STATEMENT:

-- ITS 3.7.9 Actions Note 1

Comment: There is nothing in the CTS that supports allowing Separate Condition entry for the essential and non-essential SW headers. This is a less restrictive change and the justification of A.5 is not adequate because it does not for example explain why if the headers were not treated as separate in the CTS .

NYPA RESPONSE:

The Essential and Non-Essential Service Waters headers are separate systems. This is supported by the requirement in CTS 3.3.F.4 that specifies that isolation must be maintained between the essential and nonessential headers at all times when above cold shutdown except for a period of eight hours when the headers may be cross connected while another essential header is being placed in service.

FSAR 9.6 supports this contention with the following description: The essential loads are those which must be supplied with cooling water immediately in the event of a blackout and/or Loss-of-Coolant Accident. The cooling water for these loads is supplied by the nuclear service water header. The non-essential loads are those which are supplied with cooling water from the conventional service water header. A non-essential service water pump must be manually started when required following a Loss-of- Coolant Accident.

Finally, if an essential service water pump is inoperable (three pumps required), plant operation can continue indefinitely if the essential and non-essential headers are swapped. This option demonstrates that the essential and non-essential SWS headers are separate systems and that separate condition entry is permitted by the CTS.

Based on discussions with the NRC reviewer, NYPA will revise the submittal to establish separate conditions (A and B) for an inoperable essential pump and an inoperable non-essential pump. This will allow deleting the statement that Separate condition entry is allowed for the essential and non-essential header. Editorial improvements to the wording of Conditions C and D will also be made.

NYPA revised ITS LCO 3.7.9 to split Condition A (One required SW pump on essential header inoperable; OR One required SW pump on nonessential header inoperable) into Condition A and Condition B. This eliminates the need for the Note stating that separate condition entry is allowed for the essential and non-essential headers.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.9 Service Water System (SWS)**

NRC RAI No: **3.7.9--02**

RAI STATEMENT:

-- ITS 3.7.9 Conditions A and D

Comment: Why is Condition A only pumps when the LCO is pumps and flow paths? In Condition D the purpose of the "or" after A is unclear.

NYPA RESPONSE:

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1. At IP3, there is only one essential flow path and one non-essential flow path except at the point where redundant flow paths are provided to the critical SW loads (EDGs, FCUs and CRACS). This means that all three pumps use the same flow path. If a pump is inoperable, there is a loss of redundancy but not a loss of function, and Condition A provides an AOT of 72 hours. If a redundant line to a critical SW load is inoperable, there is a loss of redundancy but not a loss of function, and Condition B or C provides an AOT of 12 hours. However, if the flow path that is common to all three pumps is inoperable, there is a loss of function and LCO 3.0.3 is applicable.

2. NYPA revised ITS so that IP3 ITS 3.7.9, Condition E, reads as follows: "Required Action and associated Completion Time of Condition A, B, C or D not met."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.9 Service Water System (SWS)**

NRC RAI No: **3.7.9--03**

RAI STATEMENT:

-- ITS SR 3.7.9.1

Comment: There is no justification for why 92 days was chosen as the frequency of the SR when the STS specifies 31 days.

NYPA RESPONSE:

CTS 3.3 and CTS 4.1 have no explicit requirements for periodic verification of SWS valve positions. IP3 is voluntarily adopting ITS SR 3.7.9.1 (see 3.7.9, DOC M.3) which requires a periodic verification of valve lineup. NYPA proposes to perform the valve lineup at a 92 day Frequency versus the 31 day Frequency specified in NUREG-1431. The 92 day Frequency was selected because it matches the frequency for swapping the essential and non-essential SW headers and this is the most likely event with a potential for a mis-positioned valve. ITS SR 3.7.9.1 is expected to be performed shortly after the header swap. NYPA believes that performing ITS SR 3.7.8.1 at a 31 day Frequency would be a significant burden without a commensurate contribution to plant safety. NYPA revised the ITS to include a justification as JFD X.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.9 Service Water System (SWS)**

NRC RAI No: **3.7.9--04**

RAI STATEMENT:

-- ITS 3.7.9 Conditions B and C

Comment: (1) It appears from review of the SW system that there are potentially other active valves not included in these Conditions that could adversely affect the system. (PIV- 1296/97 failing closed may be a problem and for PCV-1178A/1179 failing to isolate may cause a problem), (2) What is meant by Bases insert B 3.7-41-01 that the valves have been "modified"?

NYPA RESPONSE:

1. PCV 1296 and 1297 and PCV 1178A and 1179 are pressure control valves that are not actuated by an ESFAS and are not required to change position for the SWS to perform its safety function. These valves are described in Design Basis Document 304, Service Water, Section 3.7 and 3.8, which provide the following insights:

1.a PCV 1296 and 1297 are the CRAC Pressure Control Valves (one for each SW header) and are required to limit the SW pressure at the inlet to the CRAC Condensers to ensure that the design pressure of the CRAC Condensers is not exceeded. There are no accident mitigating functions and the valves are designed to fail in the open position. Recently, these valves have been removed from the system as part of a design change.

1.b PCV 1178A and 1179 are the Turbine Building Essential Header Pressure Control Valves which regulate service water pressure to the main turbine lube oil coolers, seal oil coolers, and feedpump lube oil coolers. This is required to prevent contamination of the oil by SW in the event of a tube leak. The flow coefficients for valves, PCV-1179 and PCV-1179A in full open position are less than 950 and 350, respectively. These flow coefficients limit the amount of service water to the essential Turbine Building loads during the post-LOCA injection phase concurrent with a loss of instrument air.

2. ITS LCO 3.7.9 Bases state: "Required ESFAS flow to both CRACS is provided continuously because the redundant SW to CRACS valves (TCV-1310/1311 and TCV-1312/1313) have been modified to provide the required flow at all times." Whereas the EDGs and FCUs get required service water flow when either of two valves open following an ESFAS signal, the SW supply to the CRACS does not depend on the equivalent valves opening to the CRACS because the valve internals have been removed. The intent of the statement is to communicate that these valves are now passive devices that cannot become inoperable.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.10 Ultimate Heat Sink (UHS)**

NRC RAI No: **3.7.10--01**

RAI STATEMENT:

-- STS 3.7.9.1

Comment: There is no justification provided for not adopting SR 3.7.9.1. The ITS Bases state that the UHS has a supporting structure capacity of 30 days. However, that 30 day supply is ensured by maintaining a specified UHS water level.

NYPA RESPONSE:

STS 3.7.9.1 is a bracketed requirement for periodic verification of the ultimate heat sink level with acceptance criteria designed to assure both the quantity and accessibility of required emergency cooling water in accordance with requirements in Regulatory Guide 1.27.

IP3 CTS do not include requirements for periodic verification of UHS level because the UHS is the Hudson River. FSAR Section 9.6 provides a detailed discussion of how the Hudson River satisfies requirements for both quantity and accessibility of cooling water at the IP3 site under all natural and man made conditions. FSAR 9.6 includes the following discussion:

"The service water pumps can therefore obtain water through four separate intakes, each equipped with means to prevent debris from entering the pumps and each capable of supplying all the water required for the service water pumps. Even if the main circulating pump intake were 90% blocked, that intake alone would be capable of supplying all water required for the service water pumps at design conditions. The extreme low level condition for the river at the intake structure is 4'5" below the mean sea level at the site. With the service water pump suctions at 10'-11 3/8" below the mean sea level at the site, adequate submergence at the service water pump suctions is assured."

The ITS conversion submittal did not generally include any justification for not including bracketed SRs if an equivalent requirement is not in the CTS.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.11 Control Room Ventilation System (CRVS)**

NRC RAI No: **3.7.11--01**

RAI STATEMENT:

--Bases 3.7.11

Comment: Page B 3.7.11-4, last para, should be "the" rather than "tdhe". Additionally, on page B 3.7.11-1 in BACKGROUND, third para, "Each that are... is...?

NYPA RESPONSE:

Corrected

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.11 Control Room Ventilation System (CRVS)**

NRC RAI No: **3.7.11--02**

RAI STATEMENT:

--SR 3.7.11.4

Comment: "Slightly positive pressure" Is that greater than anything greater than zero, the .125" specified for the FSB ventilation, or some other acceptance limit? In other words how will this be verified?

NYPA RESPONSE:

The methods and acceptance criteria for the surveillance test that will be incorporated into the ITS as SR 3.7.11.4 was observed within the scope of an NRC Operational Safety Team Inspection (OSTI) held June 14 to June 25, 1999. This test checks for air flow direction being out of the control room and does not quantify the differential pressure. Although the test method and acceptance criteria are not specifically discussed in NRC Engineering and Operational Safety Team Inspection Report 50-286/99-06, the methods and acceptance criteria pertaining to the verification of 'slightly positive pressure' were not called into question in the report.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.11 Control Room Ventilation System (CRVS)**

NRC RAI No: **3.7.11--03**

RAI STATEMENT:

--ITS 3.7.11

Comment: DOC M.1 states that CTS 3.3.H.1 requires that CRVS be operable but does not require redundant CRVS capability and goes on to say that CTS 3.3.H.2 establishes requirements when the CRVS is not operable but does not distinguish between a loss of CRVS redundancy and a loss of CRVS function. After reviewing the system, the CTS requirements are apparently constructed that way because the system does not have full redundancy in that a number of active components such as dampers, a single filter unit and an exhaust fan are shared. If that is the case, what is the justification for the ITS permitting 7 days to restore a train if one of two trains is inoperable?

NYPA RESPONSE:

The IP3 CRVS design is similar to the HB Robinson CREFS design. Consistent with the approach used at HB Robinson, ITS 3.7.9 maintains the CLB Allowable Out of Service Time (AOT) for a loss of CRVS function and voluntarily adopts the NUREG-1431 AOT for a loss of redundancy for those portions of the CRVS that are redundant. This is a more restrictive change.

This approach is necessary because this system was not originally designed as an emergency safeguards function that had to satisfy General Design Criteria and/or IEEE-279 requirements for redundancy and single failure tolerance. Therefore, some active components (e.g., dampers) are designed to fail safe but are not redundant. IP3 retained the 72 hour AOT for a loss of function to allow some time to correct the failure of a single component that may result in the inoperability of both trains.

Note that Zion has a similar design and elected not to adopt the NUREG-1431 AOT for a loss of redundancy for those portions of the CRVS that are redundant.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.11 Control Room Ventilation System (CRVS)**

NRC RAI No: **3.7.11--04**

RAI STATEMENT:

-- ITS 3.7.11 LCO Bases

Comment: Bases insert B 3.7-50-02 states that an inoperable CRACS fan will affect the flow balance of the CRVS due to interconnect ductwork. Given that statement how do the ITS ensure that operability under ITS 3.7.11 is maintained when that 3.7.12 component is inoperable? Additionally, the insert added the word "reticulated" to the Bases.

NYPA RESPONSE:

Flow balance is performed with Air conditioning system fan in operation. Therefore, if inoperability of AC system affects AC system fan (versus compressor, etc), then the potential exists that the inoperability of the CRACS will result in the inoperability of CRVS. Typo corrected.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.12 Control Room Air Conditioning System (CRACS)**

NRC RAI No: **3.7.12--01**

RAI STATEMENT:

--Bases 3.7.12

Comment: On page B 3.7.12-2 Applicable Safety Analysis it states that the CRACS does not satisfy all requirements in IEEE-279 for single failure. Specifically, what requirements are not met and for what components. Alternatively, where in the FSAR or other DB document is this issue elaborated on.

NYPA RESPONSE:

The control room air-conditioning system was originally designed as an industrial grade system and was not designed to meet IEEE-279

For a complete discussion of this subject, see IP3 LER 96-006, Rev. 1. In addition, the CRACS is discussed in FSAR Section 9.9.

This statement was included in the Bases because some parts of the automatic initiation and temperature control circuits are not redundant.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.12 Control Room Air Conditioning System (CRACS)**

NRC RAI No: **3.7.12--02**

RAI STATEMENT:

-- ITS 3.7.12

Comment: The CRACS appears to have the same lack of redundancy/independence discussed for the CRVS in 3.7.11-1. If that is the case, how is the 30 days for an inoperable train in the ITS justified?

NYPA RESPONSE:

See response to RAI 3.7.11-03.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.12 Control Room Air Conditioning System (CRACS)**

NRC RAI No: **3.7.12--03**

RAI STATEMENT:

-- ITS SR 3.7.12.1

Comment: The Bases of the SR says that the system is design to remove the heat load assumed in the control room. Under normal conditions, accident conditions, as specified in the FSAR or accident analysis?

NYPA RESPONSE:

NYPA revised the ITS to specify that SR 3.7.12.1 applies to the "heat load required to maintain functional capacity of the Control Room at all times (Ref. 1)." This terminology matches the description in FSAR 9.9 (i.e., Reference 1) and in the Safety Analysis section of the ITS 3.7.12 Bases.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.13 Fuel Storage Building Emergency Ventilation System (FSBEVS)**

NRC RAI No: **3.7.13--01**

RAI STATEMENT:

-- ITS SR 3.7.13.1 and DOC M.2

Comment: The frequency in the SR is 92 days, the frequency justified in the DOC is prior to moving fuel and 92 days thereafter.

NYPA RESPONSE:

DOC M.2 uses the phrase "prior to moving fuel and 92 days thereafter" to describe the application of ITS SR 3.0.1 to a Frequency of 92 days.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.15 Spent Fuel Pit Boron Concentration**

NRC RAI No: **3.7.15--01**

RAI STATEMENT:

-- ITS SR 3.7.15.1 Bases

Comment: The SR Bases refer to Figure 3.7.16-2. There is a Figure 3.7.16-1 and a Figure B 3.7.16-1 but not a Figure 3.7.16-2.

NYPA RESPONSE:

NYPA will revise the ITS conversion submittal to reference Figure B 3.7.16-1 in the Bases for SR 3.7.15.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.16 Spent Fuel Assembly Storage**

NRC RAI No: **3.7.16--01**

RAI STATEMENT:

--Bases 3.7.16

Comment: The right side of figure B 3.7.16-1 is cut off and the page title for page B 3.7.16-6 is "Assambly" instead of "Assembly".

NYPA RESPONSE:

NYPA revised the ITS to correct these discrepancies.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.7.17 Secondary Specific Activity**

NRC RAI No: **3.7.17--01**

RAI STATEMENT:

-- CTS Table 4.1.2

Comment: The second entry under Sample #6 is a Gross Activity Sample to be performed 3 times per 7 days. That requirement has not been retained nor has it been justified for relocation or deletion.

NYPA RESPONSE:

NYPA revised the ITS to delete CTS Table 4.1.2, Item 6, Part 2, Secondary Coolant Gross Activity, in ITS 3.7.17, DOC L.1. Justification for the deletion of this requirement is that the purpose of the test is early identification of SG tube leakage and that other indicators provide more timely identification of small changes in SG tube leakage.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.9.1 Boron Concentration**

NRC RAI No: **3.9--01**

RAI STATEMENT:

ITS 3.9.1 Boron Concentration

CTS 3.8.D requires verifying the boron concentration of the RCS is within specified limits every 24 hours when in Mode 6. ITS SR 3.9.1.1 requires the same verification every 72 hours under the same conditions. This is an extension of the Surveillance Test Interval from 24 hours to 72 and is therefore Beyond Scope.

NYPA RESPONSE:

Extending the Frequency for verification of the boron concentration of the RCS from 24 hours to 72 hours is justified in DOC L.1. This SR Frequency is consistent with industry practice and NUREG-1431. NYPA does not consider this a beyond scope change.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.9.1 Boron Concentration**

NRC RAI No: **3.9--02**

RAI STATEMENT:

CTS 3.8.D requires that the boron concentration in the RCS be maintained at specified levels which are detailed in the LCO. ITS 3.9.1 does not include these details but refers the reader to the COLR. Does IP-3 have an approved COLR.

NYPA RESPONSE:

Use of a COLR for IP3 was approved by Amendment 103 to the Technical Specifications. NYPA has controlled and updated the COLR for each subsequent core reload in accordance with 10CFR50.59 and the applicable Core Reload Design Change Package. The existing COLR does not identify the RCS boron concentration limit of CTS 3.8.D, but this detail will be added as part of the ITS implementing actions.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.9.2 Nuclear Instrumentation**

NRC RAI No: **3.9--03**

RAI STATEMENT:

3.9.2 Nuclear Instrumentation

CTS 3.8.A.4 requires the subcritical core is continuously monitored by two Source Range Monitors (SRM's) with one SRM that has an audible indication in containment during core alterations. ITS 3.9.2 maintains the requirements for SRM's under the same conditions but does not include the details of the audible indication in containment. DOC LA.2 states that the details are moved to the FSAR and plant procedures and also states that the audible alarm is not retained because ITS 3.9.1 provides protection for this event. Are all of the details that are being moved going to both the FSAR and Plant Procedures, or are some going to the FSAR and some to plant procedures.

NYPA RESPONSE:

NYPA is aware that requirements relocated out of CTS must be relocated to a document governed by 10 CFR 50.59.

NYPA revised DOC LA.2 of the ITS to provide the clarification that the detail related to audible indication in containment is relocated to the FSAR and "will be implemented by" plant procedures.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.9.2 Nuclear Instrumentation**

NRC RAI No: **3.9--04**

RAI STATEMENT:

ITS 3.9.2.2 has changed the Frequency from [18] to 24 months. This is an extended Surveillance Test Interval and is therefore Beyond Scope.

NYPA RESPONSE:

NYPA revised DOC M.2 of the IP3 ITS to provide the clarification that channel calibration for SRMs of 24 months when in Mode 6 is identical to the SRM calibration Frequency in ITS LCO 3.3.1 when SRMs are required in Modes 2, 3, 4 and 5.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.9.3 Containment Penetrations**

NRC RAI No: **3.9--05**

RAI STATEMENT:

ITS 3.9.3 Containment Penetrations

STS LCO 3.9.3.c requires that each penetration providing direct access from the containment atmosphere to the outside atmosphere is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System. ITS LCO 3.9.3.c.2 changes the CTS requirement by stating , capable of being closed by OPERABLE Containment Purge System Isolation Instrumentation. JFD DB.1 states That the changes to the STS are intended to improve the clarity and self explanatory. There is no technical justification for the deviation to the STS. Provide discussion and justification explaining the technical difference between Containment Purge and Exhaust Isolation Systems and Containment Purge System Isolation Instrumentation and whether or not this is your current licensing basis.

NYPA RESPONSE:

Throughout the CTS including CTS 3.8.A.8, requirements are established for the "Containment Vent and Purge System" as if is a singe system. Throughout the ITS (LCO 3.3.6, 3.6.3 and 3.9.3), NYPA breaks this into two parts: the Containment Purge System (i.e., 36-inch supply and exhaust lines per FSAR 5.3.2.3) and the Containment Vent (i.e., 10 -inch pressure relief line per FSAR 5.3.2.4) which will be referred to in the ITS as the Pressure Relief System.

In ITS 3.9.3, breaking the "Containment Vent and Purge System" into requirements for the Containment Purge System and the Containment Pressure Relief System is done in DOC M.3.

There are two changes to STS LCO 3.9.4.c.2 (IP3 LCO 3.9.3.c.2):

1. NUREG-1431, 3.9.4.c.2, refers to the "Containment Purge and Exhaust Isolation System" but the IP3 ITS refers to the "Containment Purge System" with the 3.9.3 Bases (INSERT: B 3.9-12-01) explaining that the Containment Purge System consists of 36-inch containment purge supply and exhaust lines. The supply system includes roughing filters, heating coils, fan and a containment penetration with two butterfly valves for isolation. The exhaust system includes a containment penetration with two butterfly valves for isolation and can be aligned to discharge to the atmosphere through the plant vent either directly or through the Containment Purge Filter System (i.e., a filter bank with roughing, HEPA and charcoal filters). Use of "Containment Purge System" (versus "Containment Purge and Exhaust Isolation System" in NUREG) is consistent with FSAR 5.3.2.5 and control room labeling and is used throughout the ITS (LCO 3.3.6, 3.6.3 and 3.9.3) and is explained in Bases INSERT: B 3.9-12-01 which is also notated as DB.1

2. Revised ITS LCO 3.9.4 to require that the valves are "capable of being closed by OPERABLE Containment Purge Isolation System."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.9.3 Containment Penetrations**

NRC RAI No: **3.9--06**

RAI STATEMENT:

CTS 3.8.A.8 requires testing and verifying Operable. The containment vent and purge system, including the radiation monitors which initiate isolation within 100 hours prior to refueling operation. ITS ST 3.9.3.3 performs the same surveillance but extends the Surveillance Test Interval from 100 hours to every 92 days. ITS JFD CLB.1 identifies this as your licensing basis, but CTS 3.8.A.8 DOC L.1 indicates otherwise. Extensions of Surveillance Test Intervals are Beyond Scope issues.

NYPA RESPONSE:

This change is required to implement the ITS. One of the objectives of the ITS is to eliminate the use of a Frequency such as "100 hours prior to entering the Applicability of the specification." Similar changes were made during the conversions of Vogtle, Peach Bottom, and Susquehanna. The following excerpt from the Vogtle SER is typical of the several reasons why this change is necessary.

"The Frequency of 100 hours prior to entering the Applicability of the specification has been deleted. The general rules governing the application of surveillance requirements, given by improved TS SR 3.0.4, require the surveillance to be successfully performed and current prior to entering the Applicability. The existing Frequency is essentially redundant to the general application rules which would require the surveillance to be performed within 7 days prior to entering the Applicability. Therefore, the deletion of this Frequency requirement is acceptable."

Another need for this change is the ambiguity that the SR need be completed only once prior to the start of a refueling outage or must be repeated every time refueling stops and re-starts. This is stated in a revision to DOC L.1

Note that ITS 3.9.3 JFD CLB.1 is not related to this issue.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.9.3 Containment Penetrations**

NRC RAI No: **3.9--07**

RAI STATEMENT:

CTS 3.8.A.9 states that movement of fuel in the reactor before the reactor has been subcritical for > 421 hours necessitates operation of the containment Building Vent and Purge System through the HEPA filters and charcoal absorbers. ITS 3.9.3.d.1 allows an additional course of action to isolate the Containment Purge System flow path.

DOC A.5 discusses increasing the amount of subcritical time from 421 hours to 550 hours and routing the Purge System through the filters but does not technically discuss the option to isolate the system. This change is different from both the CTS and ITS and is therefore Beyond Scope.

NYPA RESPONSE:

NYPA believes that ITS 3.9.2, DOC A.5, adequately justifies elimination of the ambiguous wording "necessitate operation" consistent with the Safety Evaluation Report (SER) for CTS Amendment 175. Therefore, this is not a beyond scope change.

ITS 3.9.3, DOC A.5 states the following:
CTS 3.8.A.9 specifies that movement of fuel in the reactor before the reactor has been subcritical for greater than 421 hours (See ITS 3.9.3, DOC A.6) will "necessitate operation" of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers. This means that containment ventilation does not have to be in operation but that it must vent through the filters when it is being used during the first 421 hours after shutdown.

ITS LCO 3.9.3.d specifies if the reactor has been subcritical for greater than 550 hours (See ITS 3.9.3, DOC A.6), then the Containment Vent and Purge System shall be either: isolated; or, aligned to discharge through the HEPA filters and charcoal adsorbers.

This change is needed because it corrects an ambiguity in CTS 3.8.A.9 which implies that the Containment Building Vent and Purge System must be in operation when moving fuel if the reactor has been subcritical for less than 421 hours. ITS LCO 3.9.3.d provides the clarification that the Containment Building Vent and Purge System must be lined up to the HEPA filters and charcoal adsorbers only if the system is in operation. This change is acceptable because it is consistent with the intent of CTS 3.8.A.9 as stated in the Safety Evaluation Report (SER) for CTS Amendment 175 which states that fuel cannot be moved until the reactor has been subcritical for > 550 hours (See ITS 3.9.3, DOC A.6) unless the containment building purge and vent system is "lined up to discharge through" both HEPA and charcoal filters. These restrictions ensure that the dose limit for a fuel handling accident (i.e., 75 rem to the thyroid at the exclusion area boundary which is 25 percent of the 10 CFR Part 100 limit of 300 rem) is met by either filtering any release from the containment or by allowing a greater decay time before fuel handling activities are permitted. This is an administrative change with no adverse impact on safety because there is no change to the existing requirement.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.9.1 Boron Concentration**

NRC RAI No: **3.9--08**

RAI STATEMENT:

STS 3.9.2 Unborated Water Source Isolation Valves

This LCO has been omitted in the ITS, Does it not apply to IP3.

NYPA RESPONSE:

IP3 CTS do not include requirements for isolation of unborated water sources during refueling because the potential dilution flow path is also used for normal and emergency makeup. Not requiring a Technical Specification for isolation of unborated water sources is supported by the IP3 FSAR Chapter 14 analysis of a boron dilution event during refueling. This event assumes that a dilution flow rate of 300 gpm is possible although unlikely. However, the RCS boron concentration must be reduced from 1900 ppm to approximately 1330 ppm before all shutdown margin is lost and reactor criticality is reached. Assuming a 300 gpm dilution, this would require more than 30 minutes. The FSAR analysis indicates this is ample time for the operator to recognize the audible high count-rate signal and isolate the reactor makeup source by closing valves and stopping the reactor makeup water pumps. Therefore, neither the CTS nor ITS need a requirement to isolate unborated water sources during refueling. Therefore, STS 3.9.2, Unborated Water Source Isolation Valves, is not included in the IP3 ITS.

ATTACHMENT III TO IPN-00-059

REVISION 1 PAGES FOR

PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

(Revision 1 pages are provided for ITS sections as outlined in Attachment I)

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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: MODES 1 and 2. RAI-01

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 3.	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% $\Delta k/k$ 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.

(continued)

BASES

BACKGROUND (continued)

Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of life (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.

NYPA

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.

(continued)

BASES (continued)

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 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.

RAI-51

(continued)

BASES

APPLICABILITY (continued)

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. An SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 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 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 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

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 PAGE	AMENDMENT FOR REV O SUBMITTAL	AMENDMENT FOR REV 1 SUBMITTAL	COMMENT
3.10-8	103, 97-118	181	No impact on ITS 3.1.2

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 Fuel 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.~~ — LA.1

Note to SR 3.1.2.1

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. — A.4

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.~~ — A.1

~~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~~

{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

{TSCR 97-118 incorporated as Amendment 181. SEE ITS 3.1.4}

R.1

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 Fuel 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% $\Delta k/k$ of predicted values at least once per 31 EFPD. No applicability is specified.

ITS 3.1.2 maintains the same requirement; however, the Applicability is specified as Modes 1 and 2. This change is acceptable because 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. 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% $\Delta k/k$ 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 3 (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 $\pm$ 1% $\Delta k/k$ of predicted values whenever the reactor is in Modes 1 and 2. This change is acceptable because 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 change will not result in a significant increase in the probability or consequences of an accident previously evaluated because the reactor is shut down in MODES 3, 4, and 5 and the reactivity balance is not changing.

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 reactor is shut down in MODES 3, 4, and 5 and the reactivity balance is not changing.

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/INCORPORATED	Removed in Rev. 1	**NA**
CEOG-058	**142 R0**	INCREASE THE COMPLETION TIME WHEN THE CORE REACTIVITY BALANCE IS NOT WITHIN LIMIT	APPROVED/INCORPORATED	Incorporated	**T.3**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	APPROVED/INCORPORATED	Incorporated	**T.1**

<CTS> 3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~3~~ Core Reactivity 2

LCO 3.1.~~3~~ The measured core reactivity shall be within ± 1% Δk/k of predicted values.
<3.10.10>

<DOC L.1> APPLICABILITY: MODES 1 and 2. R.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC L.2> 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. AND A.2 Establish appropriate operating restrictions and SRs.	72 hours 7days T.3 72 hours 7days T.3
<DOC L.2> B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours R.1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.~~3~~.1 2 <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% $\Delta k/k$ 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} > 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.

1 are PA.1

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)

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 (BOC) 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 BOC 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 BOC 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 BOC 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 ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

Insert: B 3.1-14-01

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% $\Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from

(continued)

INSERT: B 3.1-14-01

PA-1

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.

BASES

LCO (continued)

that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within 1% $\Delta k/k$ 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. R.1

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)

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~~ 7days 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~~ 7days 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% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by SR 3.1.1.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. R.1

(continued)

PA.1

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.~~3~~2.1

during physics testing following refueling

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.

during steady state operation because

also

BOL

if performed,

Insert B 3.1-17-01

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 Not Used.

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 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	**013 R1**	MOVE SR FOR 300 PPM MTC MEASUREMENT TO FREQUENCY NOTE OF SR 3.1.4.3	APPROVED/INCOR PORATED	Incorporated	**T.2**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	APPROVED/INCOR PORATED	Incorporated	**T.1**

BASES

APPLICABLE SAFETY ANALYSES (continued)

whether the reactor is at full or zero power, and whether it is the ~~BOC or EOC life~~ 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 ~~BOC and EOC~~ BOL and EOL. An EOC EOL measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC EOL value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of ~~the NRC Policy Statement~~ 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.~~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 BOL; 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.

mean | R.1

EOL

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC BOL and EOC 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 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)

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, the difference between each individual indicated rod position and its group step counter demand position shall be within the limits specified in Table 3.1.4-1 for the group step counter demand position; and

b. When THERMAL POWER is ≤ 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within 24 steps.

Am 197

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
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 3.	6 hours

(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
	<u>OR</u>		
	B.2.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
		<u>OR</u>	
	B.2.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	B.2.2	Reduce THERMAL POWER to $\leq$ 75% RTP.	2 hours
	<u>AND</u>		
	B.2.3	Verify SDM is within the limits specified in the COLR.	Once per 12 hours
	<u>AND</u>		
	B.2.4	Perform SR 3.2.1.1.	72 hours
	<u>AND</u>		
	B.2.5	Perform SR 3.2.2.1.	72 hours
	<u>AND</u>		
			(continued)

ACTIONS

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
	<u>OR</u>		
	D.1.2	Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>		
	D.2	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	------------------ NOTE ------------------ Not required to be met 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

RAI -01

Table 3.1.4-1

Maximum Permissible Rod Misalignment
(Indicated Rod Position minus Group Step Counter Demand Position)
When > 85 % RTP

Step Counter Demand Position (steps)	Maximum Permissible Deviations (IRPI Position minus Step Counter Demand Position) (steps)
≤ 212	≥ -12 and ≤ +12
213 to 225	≥ -12 and ≤ +17
226	≥ -13 and ≤ +17
227	≥ -14 and ≤ +17
228	≥ -15 and ≤ +17
229	≥ -16 and ≤ +17
≥ 230	≥ -17 and ≤ +17

Am 197

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.

(continued)

BASES

BACKGROUND (continued)

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs 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.

(continued)

BASES

BACKGROUND (continued)

The IRPI System provides an 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. 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 24 steps precludes a rod misalignment of > 22.5 inches when instrument error is considered. Additional misalignment is allowed near the fully withdrawn position because the top of the active core (approximately 225 steps) is less than the fully withdrawn position.

Am 157

APPLICABLE SAFETY ANALYSES

Control rod misalignments are analyzed in Reference 4. The acceptance criteria for addressing control rod inoperability or misalignment are that:

N/PA

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis identifies six possible modes of control rod failure and translates these failure mechanisms into eight analyzed cases. The eight cases are analyzed at full and part power conditions, and they fall into the following categories:

1. One or more rods misaligned out.
2. One or more rods misaligned in.
3. One group misaligned in.
4. One group misaligned out.
5. One group misaligned out with another group from the same cabinet misaligned in.
6. One entire bank misaligned out with the other bank from the same cabinet misaligned in.

The first six analyses are perfomed with the rods at their insertion limits. The next two analyses are for positions at other than the insertion limits.

7. All rods inserted below rod insertion limit.
8. One or more rods misaligned from all-rods-out position.

These eight conditions were applied to 248 possible cases, representing a wide variety of plant conditions involving allowable deviation below 85% RTP (± 24 steps) and above 85% RTP (±12 steps). In all cases, the resulting peaking factor increase was within required limits. Core subcriticality is assured through evaluation of shutdown margin versus rod worth for each reload cycle.

The allowable deviation increases when the rods are near their fully withdrawn limit, as shown in Table 3.1.4-1. This is due to the fact that the top of the active core is at an equivalent rod position of about 224 steps withdrawn. Therefore, the effect of increased deviation in this region is reduced for bank demand positions within 12 steps of the top of the core and higher.

N/PA

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 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 their associated group step counter demand position, the following limits are placed on individual rod positions:

(continued))

BASES

LCO (continued)

a. When THERMAL POWER is > 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within the limits specified in Table 3.1.4-1 for the group step counter demand position; and

b. When THERMAL POWER is ≤ 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within 24 steps.

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 24 steps precludes a rod misalignment of > 22.5 inches when instrument error is considered.

Am 197

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.

(continued)

BASES (continued)

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.

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 within 24 steps of their group step counter demand position, the LCO may be met by reducing reactor power to ≤ 85% RTP.

(continued)

BASES

ACTIONS

B.1 (continued)

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. A one-hour allowance for thermal stabilization of rod position instrumentation, as discussed in SR 3.1.4.1, applies when determining if a rod is misaligned.

NL-PA

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 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.

(continued)

BASES

ACTIONS (continued)

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 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:

(continued)

BASES

ACTIONS

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6 (continued)

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.

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

(continued)

BASES

ACTIONS

D.1.1 and D.1.2 (continued)

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 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 met for an individual control rod until 1 hour after

RAI-2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1 (continued)

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 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 PAGE	AMENDMENT FOR REV O SUBMITTAL	AMENDMENT FOR REV 1 SUBMITTAL	COMMENT
3.10-5	112	112	
3.10-6	181	197	Allows 24 steps Control Rod Misalignment at or below 85% RTP
3.10-7	160	197	No impact
3.10-8	181	181	
3.10-10	180	197	No impact
3.10-16	180	197	No impact
3.10-17	103	197	No impact
T 4.1-1(2)	169	200	No impact
T 4.1-3(1)	178;97-156, 98-043	200	No impact

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.
	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.
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
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 Req Act B.2.3 — M.2

Applicability Mode 1 and 2 A.4

ITS 3.1.4

A.1 A.2

3.10.5 Rod Misalignment Limitations 12 hours A.9

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

24 Amend 197 R.1

LCO 3.1.4.b

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

Req Act B.1
Req Act B 2.4, B 2.5

Figure 3.10.1 See Amend 197 R

LCO 3.1.4.a

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

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. within 2 hrs reactor power 75% A.8 L.1 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.

Superceded by Amendment 197
Changes Marked R.1
See Next page

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 24 steps. A control or shutdown rod indicating a misalignment greater than 24 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.

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 for each control or shutdown rod shall be within the limits of Figure 3.10-1. A control or shutdown rod indicating a misalignment greater than that allowed by this specification 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 less than or equal 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.

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

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 3.10.5 or fails to meet the requirements of 3.10.8.~~

Mode 1 and 2 — A.4

LCO 3.1.4 Applicability Req Act A.2 Req Act D.2 ~~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

misaligned

Req Act B.2.6 Req Act C.1 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.~~ — LA.1 — LA.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

Superceded by Amendment 177
No impact on ITS 3.1.4
See Next page

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.

3.10.7 <u>Inoperable Rod Limitations</u>

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.

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.

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 accidents listed in Table 3.10-2 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 <u>Rod Drop Time</u>

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.

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.~~

LA.2

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 Fuel 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

Amendment No. ~~28~~, ~~61~~, ~~86~~, ~~103~~, 181

ITS 3.1.4

(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) or 22.5 inches (operating at less than or equal to 85% of rated thermal power). An indicated misalignment limit of 12 steps precludes a rod misalignment of greater than 15 inches with consideration of instrumentation error, and 24 steps indicated misalignment corresponds to 22.5 inches with instrumentation error. Additional misalignment is allowed near the fully withdrawn position, since the top of the active core (approximately 225 steps) is less than the fully withdrawn position.

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.

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ITS 3.1.4

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 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.

Control rod misalignments are evaluated "as indicated by the analog rod position indicators within one hour after control rod motion." During plant startup and power escalation, the control rods are moved regularly, but not necessarily in a continuous manner. Therefore, control rod motion shall be considered to have been stopped if control rods have not been moved in the same direction as the previous control rod motion within an hour since the last control rod movement. At the end of the hour, if control rods have not been moved, then the hour hold time for evaluating control rod misalignmnet shall also be considered to have been met.

Permitted control rod misalignments (as indicated by the analog rod position indicators within one hour after control rod motion) fall into two separate categories, which are:

a) ±24 steps of the group step counter demand position (if the power level is less than or equal to 85% of rated thermal power);

b) to within the varying allowable deviations shown in Figure 3.10-1 for power level greater than 85% of rated thermal power.

A.1

ITS 3.1.4

The allowable deviation shown in Figure 3.10-1 varies as a function of bank demand position allowing for the top of active fuel ending at a control rod position of approximately 225 steps.

For group step counter demand positions greater than 212 steps withdrawn, it is acceptable for the analog rod position indicator to indicate misalignment greater than +12 steps (as indicated on Figure 3.10-1). This is due to the top of active fuel stack being at approximately 225 steps withdrawn. Indicated misalignment in the more withdrawn direction should result in the actual rod position being no lower than 201 steps withdrawn (which is within the analyzed limits). Actual control rod positions above the top of active fuel will not result in increased peaking factors for increased misalignments. Similarly, allowable negative deviation limits may increase by 1 step for every step of group step counter demand position over the top of active fuel.

For power levels less than or equal to 85% of rated thermal power the allowable deviation may increase to ±24 steps. This is due to the rate of peaking factor margin increase (as the power level decreases) being greater than the peaking factor margin loss (due to the increased control rod misalignment). This effect is described in WCAP-14668. These limits are applicable to all control rods (of all banks) over the range of 0 to 231 steps withdrawn inclusive.

The comparison of group step counter demand position and analog rod position indicator may take place at any time up to one hour after rod motion. This allows up to one hour of thermal soak time to allow the control rod drive shaft to reach a thermal equilibrium and thus present a consistent position indication. A similar time period (up to one hour after rod motion) is allowed for comparison of the bank insertion limits and the analog rod position indicators. This comparison is sufficient to verify that the control rods are above the insertion limits and thus assures the presence

A.1

of sufficient shutdown margin to satisfy the assumptions of the safety analyses. Rod position can also be confirmed via a digital voltage meter applied to the rod position control racks, in which case the operators will continue to monitor the rod position indicators on the main control board (and on the plant computer, if available and in agreement with the digital voltage meter reading) to check for deviation.

The action statements which permit limited variation from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires measurement of peaking factors (to confirm acceptability) or a restriction in thermal power; either of these restrictions provides assurance of fuel rod integrity during continued operation.

The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumption used in the accident analysis.

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

ITS 3.1.4

TABLE 3.10-2

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)

LA.1

ITS 3.1.4

Figure 3.10-1

A.15

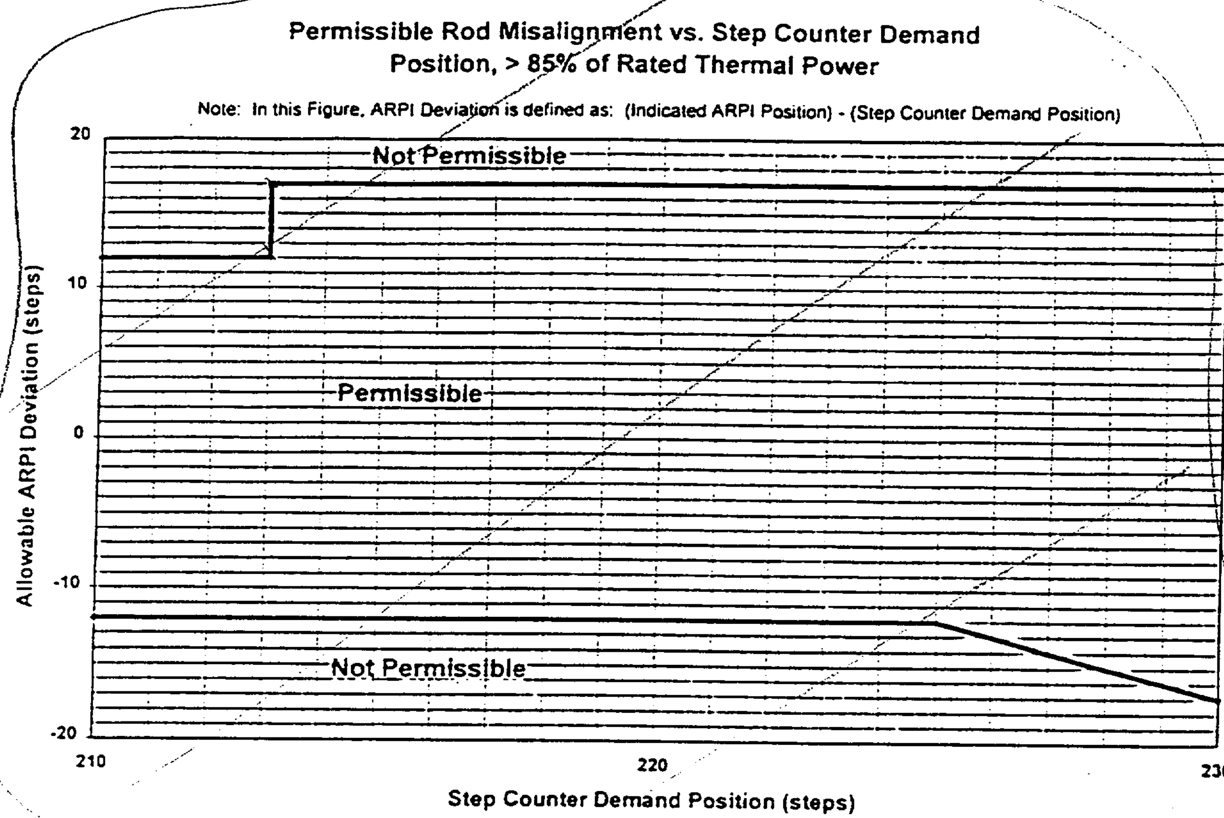

Table 3.1.4-1

Step Counter Demand Position	Maximum Deviation (ARPIs Reading ABOVE Step Counter Demand Pos)	Maximum Deviation (ARPIs Reading BELOW Step Counter Demand Pos)
≤ 212	12	-12
213	~~17~~	~~-12~~
214 - 225	17	-12
226	17	-13
227	17	-14
228	17	-15
229	17	-16
≥230	17	-17

12 hours — A9

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	
SEE CTS MASTER MARKUP	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	

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

ITS 3.1.4

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 except fully inserted rods A.11	Every 31 92 days ~~during reactor critical operations~~ L.2 A.4
	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.

Amendment No. ~~10~~, ~~14~~, ~~43~~, ~~65~~, ~~93~~, ~~99~~, ~~123~~, ~~126~~, ~~127~~, ~~129~~, ~~133~~, ~~141~~, ~~163~~, 178

Superceded by
Amends 182 & 185
No impact on ITS 3.1.4
SEE NEXT PAGE

TSCR 97-156
TSCR 98-043

ITS 3.14

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)	Quarterly
12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	24M

** 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.

SUPERCEDED BY

ITS 3.1 - R

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	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. Deleted		

** 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

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 the difference between the individual indicated rod position and its group step counter demand position is:

- within the limits specified in Table 3.1.4-1 when thermal power is >85%, and
- within 24 steps when thermal power is ≤85%.

(i.e., 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.

A.15 CTS Figure 3.10-1, added by Amendment 197, establishes limits for the maximum permissible rod misalignment (indicated rod position minus group step counter demand position) when > 85 % RTP. This information is presented in both Table and graphical format, either of which describes the LCO limits. ITS LCO 3.1.4, Table 3.1.4-1, maintains the same limits for permissible rod misalignment; however, the information is presented in Table format only and re-formatted for clarity. This is an administrative change with no impact on safety because there is no change to the limits for permissible rod misalignment.

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 nc 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 TRM. Note that TSTF-110 (WOG-49), Rev 1, deleted 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 TRM

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.

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 TRM 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
<u>("L.2" Labeled Comments/Discussions)</u>

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.

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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	**009 R1**	RELOCATE VALUE FOR SHUTDOWN MARGIN TO COLR	APPROVED/INCO RPORATED	Incorporated	**T.1**
WOG-004.2	**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	**011 R1**	DELETE "ALL" FROM LCO 3.1.5, "ROD GROUP ALIGNMENT LIMITS"	Rejected by NRC	Not Incorporated	**N/A**
WOG-004.7	**015 R1**	CORRECT ERROR IN BASES FOR LCO 3.1.5	APPROVED/INCO RPORATED	Incorporated	**T.3**
WOG-043	**107 R3**	SEPARATE CONTROL RODS THAT ARE UNTRIPPABLE VERSUS INOPERABLE	NRC Review	Not Incorporated	**N/A**
WOG-049	**110 R2**	DELETE SR FREQUENCIES BASED ON INOPERABLE ALARMS	APPROVED/INCO RPORATED	Incorporated	**T.4**

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Technical Specification 3.1.4:
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WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	APPROVED/INCORPORATED	Incorporated	**T.2**
WOG-094	**240 R0**	ELIMINATE UNNECESSARY ACTIONS TO RESTORE COMPLIANCE WITH THE LCO	TSTF Review	Not Incorporated	**N/A**
WOG-105	**314 R0**	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 <3.10.7.2> <DOC A.3> <DOC M.1>

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

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> <DOC A.6> A. One or more rod(s) untrippable.	A.1.1 Verify SDM is ≥ ~~[1.6]% Δk/k.~~ (Insert: 3.1-8-02)	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.~~ (Insert: 3.1-8-03)	1 hour
	OR	
		(continued)

T.1

T.1

Page Break

Typical

NUREG-1431 Markup Inserts
ITS SECTION 3.1.4 - Rod Group Alignment Limits

<u>INSERT: 3.1-8-01</u>

with rod group alignment limits as follows:

<3.10.5.1.b> a. When THERMAL POWER is > 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within the limits specified in ~~Figure~~ Table 3.1.4-1 for the group step counter demand position; and

<3.10.5.1.a> b. When THERMAL POWER is ≤ 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within 24 steps.

<u>INSERT: 3.1-8-02</u>

within the limits specified in the COLR. T-1

<u>INSERT: 3.1-8-03</u>

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
	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

T.1

(continued)

INSERT: 3.1-9-01

within the limits specified in the COLR.

ACTIONS (continued)

T.1

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is $\geq$ [1.6]% $\Delta k/k$.	1 hour
	<u>OR</u>	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>	
	D.2 Be in MODE 3.	6 hours

<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>

Insert: 3.1-10-02

CLB.2

SURVEILLANCE REQUIREMENTS

<DOC A.14>

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

<3.10.5.1>
<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

met

R.1

CLB.2

------------------------NOTE------------------------

<DOC A.14>

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~~.2 4 <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~~.3 4 <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

PA.1

R.1

Insert 3.1-11-01

INSERT: 3.1-11-01

Table
~~Figure~~ 3.1.4-1

Maximum Permissible Rod Misalignment
(Indicated Rod Position minus Group Step Counter Demand Position)
When > 85 % RTP

Step Counter Demand Position (steps)	Maximum Permissible Deviations (IRPI Position minus Step Counter Demand Position) (steps)
≤ 212	≥ -12 and ≤ +12
213 to 225	≥ -12 and ≤ +17
226	≥ -13 and ≤ +17
227	≥ -14 and ≤ +17
228	≥ -15 and ≤ +17
229	≥ -16 and ≤ +17
≥ 230	≥ -17 and ≤ +17

T.1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.~~5~~ 4 Rod Group Alignment Limits

BASES

i.e.

BACKGROUND

The OPERABILITY (~~e.g.~~, trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

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 ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs may ~~consists~~ consist of two groups DB.1

(continued)

Typical

BASES

BACKGROUND (continued)

that are moved in a staggered fashion, but always within one step of each other. ~~All units have~~ IP3 has four control banks and ~~at least two~~ four shutdown banks. DB.1

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~~ 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. DB.1

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are

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

DB.1

APPLICABLE SAFETY ANALYSES

Control rod misalignments accidents ~~are analyzed in the safety analysis (Ref. 3).~~ are analyzed in Reference 4. The acceptance criteria for addressing control rod inoperability or misalignment are that:

R.1

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.~~

Insert B 3.1-26-02

DB.1

~~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.~~

R.1

(continued)

INSERT: B 3.1-26-01

misalignment limit of 24 steps precludes a rod misalignment of > 22.5 inches when instrument error is considered. Additional misalignment is allowed near the fully withdrawn position because the top of the active core (approximately 225 steps) is less than the fully withdrawn position.

R.1

INSERT B 3.1-26-02

The analysis identifies six possible modes of control rod failure and translates these failure mechanisms into eight analyzed cases. The eight cases are analyzed at full and part power conditions, and they fall into the following categories:

1. One or more rods misaligned out.
2. One or more rods misaligned in.
3. One group misaligned in.
4. One group misaligned out.
5. One group misaligned out with another group from the same cabinet misaligned in.
6. One entire bank misaligned out with the other bank from the same cabinet misaligned in.

The first six analyses are perfomed with the rods at their insertion limits. The next two analyses are for positions at other than the insertion limits.

7. All rods inserted below rod insertion limit.
8. One or more rods misaligned from all-rods-out position.

These eight conditions were applied to 248 possible cases, representing a wide variety of plant conditions involving allowable deviation below 85% RTP (± 24 steps) and above 85% RTP (±12 steps). In all cases, the resulting peaking factor increase was within required limits. Core subcriticality is assured through evaluation of shutdown margin versus rod worth for each reload cycle.

The allowable deviation increases when the rods are near their fully withdrawn limit, as shown in Table 3.1.4-1. This is due to the fact that the top of the active core is at an equivalent rod position of about 224 steps withdrawn. Therefore, the effect of increased deviation in this region is reduced for bank demand positions within 12 steps of the top of the core and higher.

R.1

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).

R.1

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 their associated group step counter demand position, the following limits are placed on individual rod positions:

a. When THERMAL POWER is > 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within the limits specified in ~~Figure~~ 3.1.4-1 for the group step counter demand position; and Table

b. When THERMAL POWER is ≤ 85% RTP, the difference between each individual indicated rod position and its group step counter demand position shall be within 24 steps.

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 24 steps precludes a rod misalignment of > 22.5 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

Insert: B 3.1-28-01

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

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.

Insert B 3.1-29-02

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 within 24 steps of their group step counter demand position, the LCO may be met by reducing reactor power to ≤ 85% RTP. R.1

INSERT B 3.1-29-02

A one-hour allowance for thermal stabilization of rod position instrumentation, as discussed in SR 3.1.4.1, applies when determining if a rod is misaligned.

R.1

INSERT: B 3.1-26-01

misalignment limit of 24 steps precludes a rod misalignment of > 22.5 inches when instrument error is considered. Additional misalignment is allowed near the fully withdrawn position because the top of the active core (approximately 225 steps) is less than the fully withdrawn position.

R.1

INSERT B 3.1-26-02

The analysis identifies six possible modes of control rod failure and translates these failure mechanisms into eight analyzed cases. The eight cases are analyzed at full and part power conditions, and they fall into the following categories:

1. One or more rods misaligned out.
2. One or more rods misaligned in.
3. One group misaligned in.
4. One group misaligned out.
5. One group misaligned out with another group from the same cabinet misaligned in.
6. One entire bank misaligned out with the other bank from the same cabinet misaligned in.

The first six analyses are perfomed with the rods at their insertion limits. The next two analyses are for positions at other than the insertion limits.

7. All rods inserted below rod insertion limit.
8. One or more rods misaligned from all-rods-out position.

These eight conditions were applied to 248 possible cases, representing a wide variety of plant conditions involving allowable deviation below 85% RTP (± 24 steps) and above 85% RTP (±12 steps). In all cases, the resulting peaking factor increase was within required limits. Core subcriticality is assured through evaluation of shutdown margin versus rod worth for each reload cycle.

The allowable deviation increases when the rods are near their fully withdrawn limit, as shown in Table 3.1.4-1. This is due to the fact that the top of the active core is at an equivalent rod position of about 224 steps withdrawn. Therefore, the effect of increased deviation in this region is reduced for bank demand positions within 12 steps of the top of the core and higher.

R.1

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

T.3

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.

for

D.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

T.3

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

(continued)

BASES

ACTIONS

~~D.1 (continued)~~

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

SURVEILLANCE REQUIREMENTS

SR 3.1.~~8~~ 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 B 3.1-32-01

Insert: B 3.1-32-02

CLB.2

SR 3.1.~~8~~ 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 3.1.5.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.5.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

PA.1

Insert: B 3.1-32-03

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~~. 197

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 met 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~~ 197

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 3.1.7:
"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.	Once per 8 hours
	<u>OR</u>		
	A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

RAI-5

(continued)

ACTIONS (continued)

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.	4 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 ≤ 24 steps apart when ≤ 85% RTP.	Once per 8 hours
	OR		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

RAI-6

Am 197

(continued)

ACTIONS (continued)

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 criticality 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.

(continued)

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 24 steps precludes a rod misalignment of > 22.5 inches when instrument error is considered.

Am 197

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 orshutdown rods operating outside their

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

(continued)

BASES

LCO (continued)	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. <u>A.1</u> 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.

(continued)

BASES

ACTIONS

A.1 (continued)

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.

RAI 5

Note that an IRPI channel is not inoperable if rod position can be determined using a digital voltmeter 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 determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at ≥ 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

RAI-6

(continued)

BASES

ACTIONS (continued)

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 ≤ 24 steps apart when ≤ 85% RTP within the allowed Completion Time of once every 8 hours is adequate.

Am 197

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 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.

NYPA

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 (continued)

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 PAGE	AMENDMENT FOR REV O SUBMITTAL	AMENDMENT FOR REV 1 SUBMITTAL	COMMENT
3.10-6	181	197	Revised proposed ITS 3.1.7 to incorporate new limits for rod deviation per Amendment 197.
3.10-10	180	197	
3.10-16	180	197	
T 4.1-1(2)	169	200	Deleted Boric Acid Tank Level Surveillance. No impact on ITS 3.1.7

(A.1) (A.2) ITS 3.1.7

3.10.5 Rod Misalignment Limitations

SEE ITS 3.1.4

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.

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 4 hours (M.2) (R.1)

Req Act B.1

Req Act A.2, B.2 b. During operation below 50 percent of rating, no special monitoring is required. (L.1)

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. (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)

Supercided by Amendment 197. Changes to Req Action C.1.2

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 24 steps. A control or shutdown rod indicating a misalignment greater than 24 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.

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 for each control or shutdown rod shall be within the limits of Figure 3.10-1. A control or shutdown rod indicating a misalignment greater than that allowed by this specification 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 less than or equal 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.

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

ITS 3.1.7
Rev 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) or 22.5 inches (operating at less than or equal to 85 % of rated thermal power). An indicated misalignment limit of 12 steps precludes a rod misalignment of greater than 15 inches with consideration of instrumentation error, and 24 steps indicated misalignment corresponds to 22.5 inches with instrumentation error. Additional misalignment is allowed near the fully withdrawn position, since the top of the active core (approximately 225 steps) is less than the fully withdrawn position.

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

ITS 3.1.7 Rev. 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 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.

Control rod misalignments are evaluated "as indicated by the analog rod position indicators within one hour after control rod motion." During plant startup and power escalation, the control rods are moved regularly, but not necessarily in a continuous manner. Therefore, control rod motion shall be considered to have been stopped if control rods have not been moved in the same direction as the previous control rod motion within an hour since the last control rod movement. At the end of the hour, if control rods have not been moved, then the hour hold time for evaluating control rod misalignmnet shall also be considered to have been met.

Permitted control rod misalignments (as indicated by the analog rod position indicators within one hour after control rod motion) fall into two separate categories, which are:

a) ±24 steps of the group step counter demand position (if the power level is less than or equal to 85% of rated thermal power);

b) to within the varying allowable deviations shown in Figure 3.10-1 for power level greater than 85% of rated thermal power.

A1

ITS 3.1.7

A.7

M.1

SEE ITS 3.1.4

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
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	

SR 3.1.7.1

SEE CTS MASTER MARKUP

Amendment No. ~~8, 38, 63, 68, 74, 93, 107, 125, 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.

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 4 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 4 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 24 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 24 steps of the group step counter demand position when ≤ 85% RTP. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None

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 4 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 24 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 24 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 24 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	**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/INCORPORATED	Incorporated	**T.2**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	APPROVED/INCORPORATED	Incorporated	**T.1**
WOG-073	**234 R1**	ADD ACTION FOR MORE THAN ONE DRPI INOPERABLE	~~TSTF Review~~ Approved	Not Incorporated	**N/A**

T.1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~8~~ 7 Rod Position Indication

Individual

I

<3.10.6>
<DOC L.2>

LCO 3.1.~~8~~ 7 The ~~[Digital]~~ Rod Position Indication ([~~D~~]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 A3> <DOC M.2>	A. One [D]RPI per group inoperable for one or more groups. (I)	A.1 Verify the position of the rods with inoperable position indicators by using movable incore detectors.	Once per 8 hours	R.1
		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.	~~[4]~~ hours	R.1
		OR		

(continued)

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 ~~D~~ I RPIs for the affected banks are OPERABLE.	Once per 8 hours	
		AND			
	when > 85% RTP and ≤ 24 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	R.1 CLB.1 See Amend 197
		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
⟨Table 4.1-1 # 9⟩ ⟨DOC M.1⟩ 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]~~ Insert: 3.1-19-01 T.2

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~~ Individual Rod Position Indication (~~D~~ I RPI) 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.

I

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.~~6~~ 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 24 steps precludes a rod misalignment of > 22.5 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.

The ~~control~~ rod position indicator channels satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36. 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 DRPI I 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 DRPI System indicates within ~~12~~ the required number of steps of the group step counter demand position as required by LCO 3.1.~~5~~ 4, "Rod Group Alignment Limits";

b. For the DRPI I 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 DRPI I System.

The ~~12 step~~ agreement limit I between the Bank Demand Position Indication System and the DRPI 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. 1R.1

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, LCO 3.1.6, and LCO 3.1.7), 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 can be determined using a digital voltmeter in lieu of the installed indicators.

R. 1

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 ~~[4]~~ hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of ~~[4]~~ hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2 (i.e., bank demand position cannot be determined)

With one demand position indicator per bank inoperable (I), 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 ≤ 24 steps apart when ≤ 85% RTP within the allowed Completion Time of once every 8 hours is adequate.

R.1

(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 ~~D~~RPI I agrees with the demand position within ~~[12]~~ the required number of steps ensures that the ~~D~~RPI is operating correctly. ~~Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.~~

R.1

Insert: B 3.1-51-01

~~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: B 3.1-51-02

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) – $T_{avg} \leq$ 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. 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 3.2.1:
"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 after each $F_Q(Z)$ determination
	<u>AND</u>		
	A.2	Reduce Power Range Neutron Flux−High trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	72 hours after each $F_Q(Z)$ determination
	<u>AND</u>		
	A.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds limit.	72 hours after each $F_Q(Z)$ determination
	<u>AND</u>		
	A.4	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. 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.

(continued)

BASES (continued)

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.

(continued)

BASES (continued)

LCO

The Heat Flux Hot Channel Factor, $F_Q(Z)$, shall be limited by the following relationships:

$$F_Q(Z) \le \frac{FQ}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \le \frac{FQ}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

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

(continued)

BASES

LCO (continued)

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. 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. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q(Z)$ and would require power reductions within 15 minutes of the $F_Q(Z)$ determination, if necessary, to comply with the decreased maximum allowable power level. Decreases in the $F_Q(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

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

(continued)

BASES

ACTIONS

A.2 (continued)

preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1. The maximum allowable Power Range Neutron Flux - High trip setpoints initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q(Z)$ and would require reductions for the Power Range Neutron Flux - High trip setpoints within 72 hours of the $F_Q(Z)$ determination, if necessary, to comply with the decreased maximum allowable power level. Decreases in the $F_Q(Z)$ would allow increasing the Power Range Neutron Flux - High trip setpoints.

A.3

Reduction in the Overpower ΔT trip setpoints by $\geq$ 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. The maximum allowable Overpower ΔT trip setpoints initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q(Z)$ and would require reductions for the Overpower ΔT setpoints within 72 hours of the $F_Q(Z)$ determination, if necessary, to comply with the decreased maximum allowable power level. Decreases in the $F_Q(Z)$ would allow increasing the Overpower ΔT trip setpoints.

A.4

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

(continued)

BASES

ACTIONS

B.1. (continued)

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 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 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"

PART 2:

CURRENT TECHNICAL SPECIFICATION PAGES

Annotated to show differences between CTS and ITS

CTS PAGE	AMENDMENT FOR REV O SUBMITTAL	AMENDMENT FOR REV 1 SUBMITTAL	COMMENT
3.10-1	112	112	
3.10-2	112	112	
3.10-8	181	181	
3.10-8a	103	103	
3.10-9	175	175	
3.10-10	180	197	No impact on ITS 3.2.1
F3.10-1	112	197	No impact on ITS 3.2.1
F3.10-2	143	143	
F3.10-3	14	14	

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% $\Delta k/k$.

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

LCO 3.2.1

~~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: shall be within limits in COLR A.4 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

Amendment No. ~~23~~, ~~40~~, ~~48~~, ~~61~~, 73, ~~86~~, ~~103~~, 112

M.1

Add Note to Surveillance

~~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

Prior to > 75% RTP AND 12 hrs after pwr increase M.1

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

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.~~ A.9

M.2 within 15 min

L.1 within 72 hours

Req Act A.1
Req Act A.2

A.L

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.4 — A.7 | R.1

Add Req Action B.1 — A.8

Amendment No. ~~73~~, ~~86~~, ~~103~~, 112

Add Req Action A3 — M.3 | R.1

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 Fuel 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) or 22.5 inches (operating at less than or equal to 85 % of rated thermal power). An indicated misalignment limit of 12 steps precludes a rod misalignment of greater than 15 inches with consideration of instrumentation error, and 24 steps indicated misalignment corresponds to 22.5 inches with instrumentation error. Additional misalignment is allowed near the fully withdrawn position, since the top of the active core (approximately 225 steps) is less than the fully withdrawn position.

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

A.1

~~FIGURE 3.10-1~~

~~HAS BEEN~~

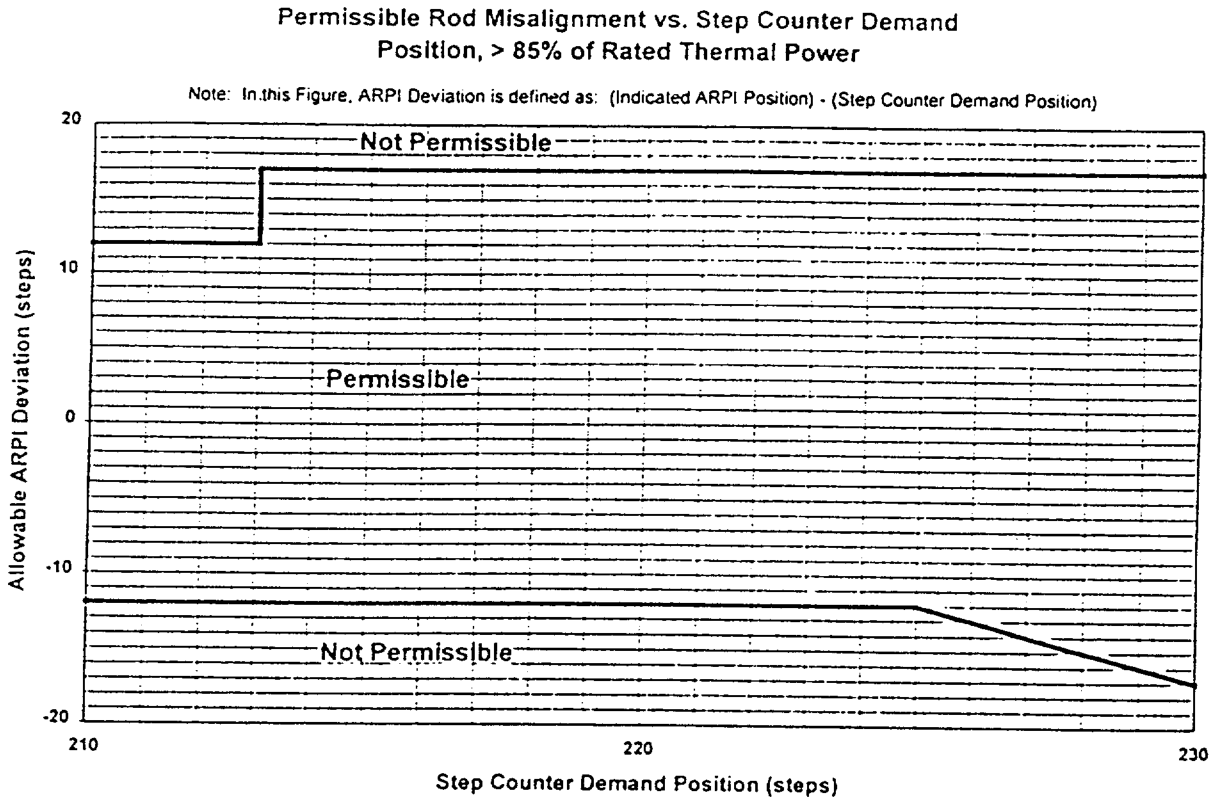
~~DELETED~~

Amendment No. 112 See next page, Amendment 197

No impact on ITS 3.2.1

R.1

Figure 3.10-1



Step Counter Demand Position	Maximum Deviation (ARPIs Reading ABOVE Step Counter Demand Pos)	Maximum Deviation (ARPIs Reading BELOW Step Counter Demand Pos)
≤ 212	12	-12
213	17	-12
214 - 225	17	-12
226	17	-13
227	17	-14
228	17	-15
229	17	-16
≥230	17	-17

Amendment No. 112, 197

No impact on ITS 3.2.1

See ITS 3.1.4

DELETED

Figure 3.10-2 ~~Hot~~ Channel ~~Factor~~ Normalized ~~Operating~~ ~~Envelope~~

A.1

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, A.2 and A.3 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.4, specifies that reactor power, Overpower ΔT trip setpoint, 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.4 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 after each $F_Q(Z)$ determination 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 after each $F_Q(Z)$ determination 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.

M.3 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 requirement for a proportional reduction of the Overpower ΔT trip setpoints.

ITS LCO 3.2.1, Required Action A.3, is added to require a reduction in the Overpower ΔT trip setpoints by ≥ 1% for each 1% by which FQ(Z) exceeds its limit within 72 hours of each determination that limits for $F_Q(Z)$ are not met. Following this setpoint reduction, continued plant operation is permitted. A proportional reduction Overpower ΔT trip setpoints when limits for $F_Q(Z)$ are not met 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. 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 after each $F_Q(Z)$ determination 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 after each $F_Q(Z)$ determination 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.

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)
Conversion Package

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/INCOR PORATED	Incorporated	**T.1**
WOG-025	**097 R0**	REVISE NOTE TO SR 3.2.1.2, FQ MEASUREMENT	APPROVED/INCOR PORATED	SR 3.2.1.2 not incorporated.	**N/A**
WOG-026	**098 R2**	RELOCATE THE FQ(Z) PENALTY FACTOR TO THE COLR	APPROVED/INCOR PORATED	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/INCOR PORATED	Incorporated	**T.3**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	APPROVED/INCOR PORATED	Incorporated	**T.2**

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.1:
"Heat Flux Hot Channel Factor (FQ(Z))"

WOG-095	**241 R4**	ALLOW TIME FOR STABILIZATION AFTER REDUCING POWER DUE TO QPTR OUT OF LIMIT	Approved/Not in Orig	Incorporated	**T.4**
WOG-101		CLARIFY COMPLETION TIME AND FREQUENCY WORDING	TSTF Review	Not Incorporated	**N/A**

3.2 POWER DISTRIBUTION LIMITS

3.2.1~~B~~ Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

⟨3.10.2.1⟩
⟨DOC LA.1⟩

LCO 3.2.1~~B~~ $F_Q(Z)$~~, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$,~~ shall be within the limits specified in the COLR. DB.1

TSTF-241,R1

⟨3.10.2.1⟩
⟨DOC A.3⟩
⟨DOC A.4⟩
⟨DOC A.5⟩

APPLICABILITY: MODE 1.

after each F_Q determination

R.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q^{\cancel{C}}(Z)$ not within limit.	A.1	Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^{\cancel{C}}(Z)$ exceeds limit.	15 minutes
	AND		
	A.2	Reduce Power Range Neutron Flux—High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	~~8~~ 72 hours
	AND		
	A.3	Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.	72 hours
	AND		
	A.4	Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

⟨3.10.2.2.2⟩
⟨DOC M.2⟩

T.1

⟨3.10.2.2.2⟩
⟨DOC L.1⟩

⟨DOC M.3⟩

R.1

⟨DOC A.7⟩

(continued)

ACTIONS (continued)

DB.1

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. $F_Q^W(Z)$ not within limits.~~	~~B.1 Reduce AFD limits $\geq$ 1% for each 1% $F_Q^W(Z)$ exceeds limit.~~	~~2 hours~~ 4 T.3
<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.

<3.10.2.2>
<DOC M.1>
<DOC A.7>

SURVEILLANCE	FREQUENCY
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~~B~~ 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.~~7~~, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis. 6

$F_Q(Z)$ 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

Insert: B3.2-2-01

DE 2

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~~. PA.1

10 CFR 50.36

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

INSERT: B 3.2-2-01

DB2

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~~. PA1

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 $F_Q(Z)$. DB1

An $F_Q^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_Q^M(Z)$) of $F_Q(Z)$. Then,

$$F_Q^C(Z) = F_Q^M(Z)\ [1.0815]$$

where [1.0815] is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

Insert: B3.2-13-01

$F_Q^C(Z)$ 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. DB1

(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:~~

$$\cancel{F_Q^W(Z) = F_Q^C(Z)\ W(Z)}$$

~~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 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 $F_Q(Z)$ 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

Insert B 3.2-14-01 T.4 R.1

~~Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^C(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.~~

(continued)

INSERT: B 3.2-14-01

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of $F_Q(Z)$ and would require power reductions within 15 minutes of the $F_Q(Z)$ determination, if necessary, to comply with the decreased maximum allowable power level. Decreases in the $F_Q(Z)$ would allow increasing the maximum allowable power level and increasing power up to this revised limit.

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. T.1 72

Insert B 3.2-15-01 T.4

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. R.1

Insert B 3.2-15-2 T.4

A.4

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. CLB1

(continued)

INSERT: B 3.2-15-01

The maximum allowable Power Range Neutron Flux - High allowable values initially determined by Required Action A.2 may be affected by subsequent determinations of $F_Q(Z)$ and would require reductions for the Power Range Neutron Flux - High allowable value within 72 hours of the $F_Q(Z)$ determination, if necessary, to comply with the decreased maximum allowable power level. Decreases in the $F_Q(Z)$ would allow increasing the Power Range Neutron Flux - High allowable value.

INSERT: B 3.2-15-02

The maximum allowable Overpower ΔT allowable value initially determined by Required Action A.3 may be affected by subsequent determinations of $F_Q(Z)$ and would require reductions for the Overpower ΔT allowable values allowable value within 72 hours of the $F_Q(Z)$ determination, if necessary, to comply with the decreased maximum allowable power level. Decreases in the $F_Q(Z)$ would allow increasing the Overpower ΔT allowable value.

BASES

ACTIONS (continued)

B
~~C~~.1

3

If Required Actions A.1 through A.~~4~~ ~~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

is

SR 3.2.1.1 ~~and SR 3.2.1.2 are~~ modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^C(Z)$ ~~and $F_Q^W(Z)$~~ ~~are~~ 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 $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z)$ [1.0815] (Ref. 4). $F_Q^C(Z)$ is then compared to its specified limits.

The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Insert: B 3.2-17-01

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met when RTP is achieved, because the highest peaking factors, generally decrease as Core average power level is increased.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required [12] hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

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)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.1 - Heat Flux Hot Channel Factor ($F_Q(Z)$)

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

$$\text{maximum over z} \quad \left[\frac{F_Q^C(Z)}{K(Z)} \right],$$

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)

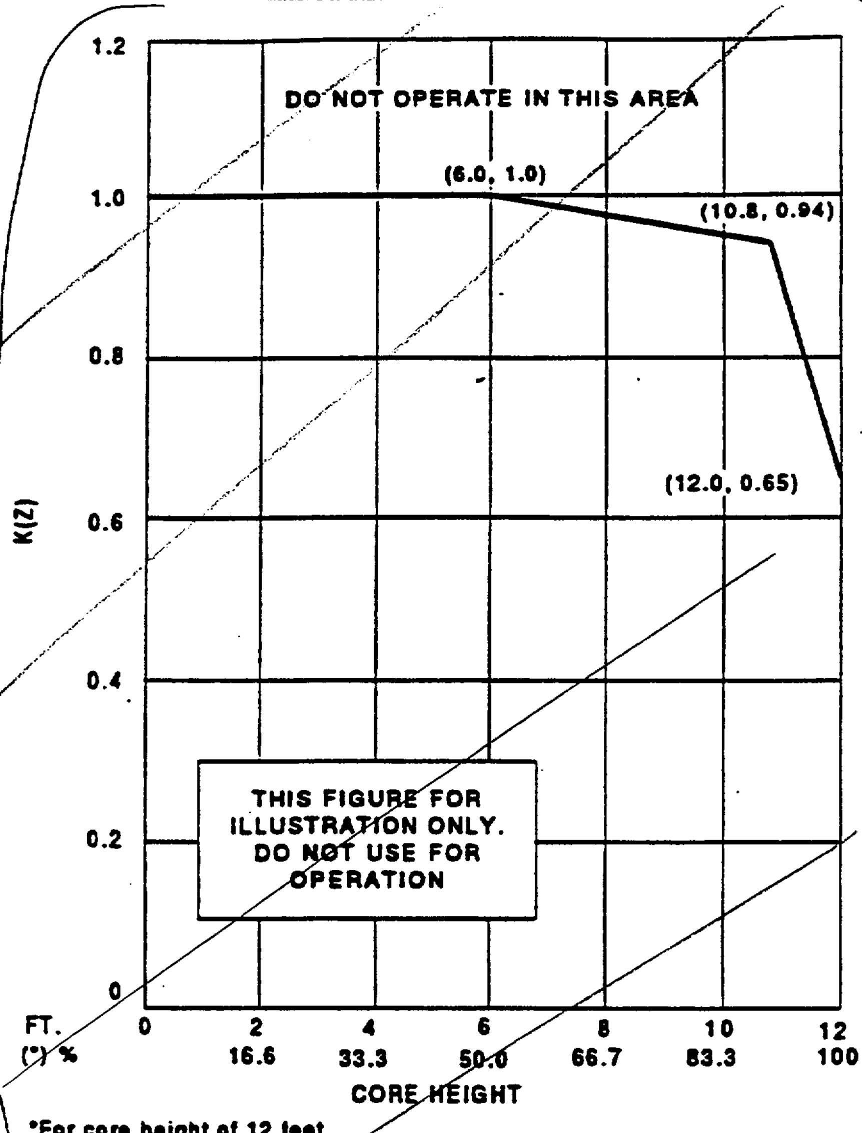


Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

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. Specifically, IP3 did not incorporate Condition B ($F_Q(Z)$ not within limits) because this requirement and the associated Action is not currently required at IP3 and applies only to plants that use Relaxed Axial Offset Control operation. IP3 uses Constant Axial Offset Control operation.

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.

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.