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New York Power Authority

James Knubel
Senior Vice President and
Chief Nuclear Officer

September 14, 2000
IPN-00-069

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, G. Wunder to J. Knubel, dated July 9, 1999, "Request for Additional Information Regarding STS Conversion."

2. NYPA letter, J. Knubel to USNRC, dated December 15, 1998 (IPN-98-139), "Proposed Technical Specification Change Conversion to ITS."

3. NYPA letter, J. Knubel to NRC, dated August 16, 2000 (IPN-00-059), Proposed ITS - Reply to NRC RAI."

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.

3.4 Reactor Coolant System (consists of 16 subsections)
3.5 Emergency Core Cooling Systems (consists of 4 subsections)
3.8 Electrical Power Systems (consists of 10 subsections)
4.0 Design Features (consists of 3 subsections)
5.0 Administrative Controls (consists of 7 subsections)

A001

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

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.

Similar information for other ITS sections was transmitted in Reference 3 and responses to RAIs for the remaining two sections (3.3 and 3.6) will be submitted by September 25, 2000.

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
Chief Nuclear Officer

STATE OF NEW YORK
COUNTY OF WESTCHESTER
Subscribed and sworn to before me
this 14th day of Sept. 2000.

EILEEN E. O'CONNOR
Notary Public, State of New York
No. 4991062
Qualified in Westchester County
Commission Expires January 21, 2002

cc: Next page

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REVISION STATUS FOR PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

ITS NUM	ITS SECTION TITLE	NRC RAIS	New Amendment	NYPA Changes	COMMENT
3.4	REACTOR COOLANT SYSTEM	(58)			
3.4.1	RCS P,T, and Flow DNB Limits	4	Yes See comment	No	Changes reflect response to RAIs and DOC A.6 superseded by Amendment 191. Affected Revision 1 pages submitted for review.
3.4.2	RCS Min Temp for Criticality	1	Yes No impact	No	Typographical error corrected per RAI. Submittal of Revision 1 proposed ITS not required.
3.4.3	RCS P/T Limits	6	No	Yes	Changes reflect response to RAIs. NYPA changes reflected in the Bases. Revision 1 of proposed ITS submitted for review.
3.4.4	RCS Loops Mode 1 & 2	2	Yes No impact	No	A-DOCs reclassified as M-DOCs per RAI. Submittal of Revision 1 of proposed ITS not required.
3.4.5	RCS Loops Mode 3	5	No	No	Changes reflect response to RAIs. Affected Revision 1 pages submitted for review.
3.4.6	RCS Loops Mode 4	4	No	Yes	Changes reflect responses to RAIs and NYPA changes to the Bases. Affected Revision 1 pages submitted for review.
3.4.7	RCS Loops Mode 5, filled	3	No	Yes	Changes reflect responses to RAIs. NYPA changes reflected in the Bases. Affected Revision 1 pages submitted for review.
3.4.8	RCS Loops Mode 5, not filled	1	No	Yes	Change reflects response to RAI. NYPA changes reflected in Specification and Bases. Affected pages submitted for review.
3.4.9	Pressurizer	3	No	No	Changes reflect response to RAIs. Revision 1 of proposed ITS submitted for review.
3.4.10	Pressurizer Safety Valves	3	Yes No impact	No	Changes reflect response to RAIs. Affected Revision 1 pages submitted for review.
3.4.11	Pressurizer PORVs	9	No	No	Changes reflect response to RAIs. Affected Revision 1 pages submitted for review.
3.4.12	LTOP	2	Yes No impact	Yes	Changes reflect response to RAIs. Affected Revision 1 pages submitted for review.
3.4.13	RCS Operational Leakage	2	Yes No impact	No	Changes reflect response to RAIs. Affected Revision 1 pages submitted for review.
3.4.14	RCS Pressure Iso. Valve Lkage	8	No	Yes	Changes reflect response to RAIs. NYPA changes reflected in Specification and Bases. Revision 1 of proposed ITS submitted.
3.4.15	RCS Leak Detection Instr.	2	Yes No impact	No	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
3.4.16	RCS Specific Activity	3	Yes No impact	Yes	Changes reflect response to RAIs. NYPA changes reflected in CTS markup and DOCs. Affected pages submitted for review.

REVISION STATUS FOR PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

ITS NUM	ITS SECTION TITLE	NRC RAIs	New Amendment	NYPA Changes	COMMENT
3.5	Emergency Core Cooling Systems	5			
3.5.1	Accumulators	0	Yes No impact	Yes	NYPA changes reflect changes to the bases. Affected Revision 1 pages submitted for review.
3.5.2	ECCS - Operating	2	Yes see comment	No	Changes reflect response to RAIs and incorporate amendment 196. Affected Revision 1 pages submitted for review.
3.5.3	ECCS - Shutdown	1	No	Yes	Changes reflect response to RAIs. NYPA changes reflect changes to DOCs, specification and bases. Affected Revision 1 pages submitted for review.
3.5.4	Refueling Water Storage Tank	2	Yes No impact	Yes	Changes to Spec and Bases reflect response to RAIs and NYPA addition of RWST level instrumentation. Affected Revision 1 pages submitted for review.
3.8	ELECTRICAL POWER SYSTEMS	(49)			
3.8.1	AC Sources - Operating	19	Yes see comment	Yes	Spec and Bases revised to reflect reply to RAIs and NYPA changes. Bases change from Amendment 201 included in ITS Bases. Affected Revision 1 pages submitted for review.
3.8.2	AC Sources - Shutdown	6	Yes See comment	No	Spec and Bases revised to reflect reply to RAIs. Amendment 194 incorporated to allow 1 DG in CSD.
3.8.3	DG Fuel Oil and Starting Air	10	Yes See comment	Yes	Spec and Bases revised to reflect reply to RAIs. NYPA change to oil volume requirement reflected in Spec, Bases, and DOCs. Also, Amendment 194 (see ITS 3.8.2) resulted in change to Bases. Affected Revision 1 pages submitted for review.
3.8.4	DC Sources - Operating	8	Yes No impact	Yes	Spec and Bases revised to reflect reply to RAIs. Two new changes identified by NYPA are shown in the Bases. Affected Revision 1 pages submitted for review.
3.8.5	DC Sources - Shutdown	2	No	None	Bases revised to reflect replies to RAIs. Affected Revision 1 pages submitted for review.
3.8.6	Battery Cell Parameters	0	Yes No impact	Yes	Affected Revision 1 pages submitted for review of two new changes proposed by NYPA.
3.8.7	Inverters - Operating	1	Yes No impact	None	Changes made in 3.8.7 to address the RAI are identical to changes made in 3.8.8. Refer to 3.8.8 revision pages.
3.8.8	Inverters - Shutdown	2	No	None	Spec and Bases revised to reflect reply to RAI. Affected Revision 1 pages submitted for review.
3.8.9	Distribution Sys - Operating	1	Yes No impact	None	Bases changed to reflect reply to RAI. Affected Revision 1 pages submitted for review.
3.8.10	Distribution Sys - Shutdown	0	No	None	No changes to ITS, Rev 0. Submittal of Rev. 1 not required.

REVISION STATUS FOR PROPOSED IMPROVED TECHNICAL SPECIFICATIONS

ITS NUM	ITS SECTION TITLE	NRC RAIs	New Amendment	NYPA Changes	COMMENT
4.0	DESIGN FEATURES	(3)	Yes No impact	None	Changes to ITS, DOCs and JFD per RAI response. Affected Revision 1 pages submitted for review.

ITS NUM	ITS SECTION TITLE	NRC RAIs	New Amendment	NYPA Changes	COMMENT
5.0	ADMINISTRATIVE CONTROLS	8			
5.1	Responsibility	0	Yes No impact	Yes	Relocation destination for DOC LA.1 changed from 'QA Plan' to 'FSAR'. Submittal of Revision 1 proposed ITS not required.
5.2	Organization	0	Yes see comment	None	Amendment 193 removed a temporary note pertaining to CTS 6.2.2(i). DOC A.5 is no longer required and is deleted for ITS Revision 1. No change to Revision 0 Spec or Bases. Submittal of Revision 1 proposed ITS not required.
5.3	Unit Staff Qualifications	0	No	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
5.4	Procedures	0	Yes No impact	None	No changes to Revision 0 proposed ITS. Submittal of Revision 1 proposed ITS not required.
5.5	Programs and Manuals	8	Yes No impact	Yes	Revised DOCs in 5.5.8, added CLB notation in 5.5.9, revised certain test frequencies in 5.5.10, and changed oil testing requirements in 5.5.12 per RAIs. Affected Revision 1 pages submitted for review.
5.6	Reporting Requirements	0	Yes No impact	Yes	Updated to delete PTLR per revision 1 of ITS 1.0 and added COLR contents consistent with ITS 3.1.1, 3.1.3, and 3.9.1. Affected Revision 1 pages submitted for review
5.7	High Radiation Area	0	Yes No impact	Yes	Restored STS alternatives to control of very high radiation areas. Revision 1 of proposed ITS submitted for review.

ATTACHMENT II TO IPN-00-069

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
3.4	Reactor Coolant System	58
3.5	Emergency Core Cooling Systems	5
3.8	Electrical Power Systems	49
4.0	Design Features	3
5.0	Administrative Controls	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: 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

NRC RAI No: **3.4 -- 00**

RAI STATEMENT:

A.1 Docs ---Some of the detail that is shown as deleted has actually been moved to the Bases.

NYPA RESPONSE:

DOC A.1 states the following:
"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."

NYPA is not aware of any requirement to or any benefit from preparing a markup the CTS Bases to show that specific information was retained in the ITS Bases. The percentage of information in the CTS Bases that was retained in the ITS Bases is very small and DOC A.1 does not prohibit use of CTS Bases information in the ITS Bases. Additionally, the SER associated with each CTS Amendment is considered the source of CTS information retained in the ITS.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

NRC RAI No: **3.4 -- 01**

RAI STATEMENT:

DOC A.4 indicates that the requirement to verify SLs are met are not included in LCO 3.4.1 because ITS SL 2.1.1, Reactor Core SLs are less restrictive than the limits in ITS 3.4.1.

Comment: Please clarify what is meant by this statement.

NYPA RESPONSE:

ITS SL 2.1.1, Reactor Core SLs, establishes limits for the combination of THERMAL POWER, Reactor Vessel inlet temperature, and pressurizer pressure and Required Actions if these limits are exceeded. ITS LCO 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, establishes limits for pressurizer pressure, RCS average temperature, and RCS total flow rate that ensure that the SLs in ITS SL 2.1.1 are not exceeded. Therefore, ITS LCO 3.4.1 limits will be exceeded before ITS SL 2.1.1 are violated. However, both CTS 3.1.H.4 and the Bases for LCO 3.4.1 specify that if ITS LCO 3.4.1 limits are exceeded, then operators should determine if limits in SL 2.1.1 have been exceeded to determine if the more stringent Required Actions associated with SL 2.1.1 must be implemented.

Moving a cross reference that advises operators to ensure that another Technical Specification has not been exceeded from the body of the CTS to the ITS Bases is an administrative change because both the CTS 2.0 and ITS 2.1.1 requirement not to exceed safety limits at any time is unaffected by elimination of a cross reference between specifications.

Minor editorial improvements were made to 3.4.1 DOC A.4 to improve clarity.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

NRC RAI No: **3.4 -- 02**

RAI STATEMENT:

DOC A.6 refers to a footnote that has a commitment for NRC review and approval. The STS should not delete this requirement without NRC review or some indication that the commitment is no longer required. Beyond Scope.

NYPA RESPONSE:

The footnote associated with CTS 3.1.H, RCS Pressure, Temperature and Flow DNB Limits, was deleted by CTS Amendment 191. The IP3 ITS conversion submittal was revised to incorporate CTS Amendment 191.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

NRC RAI No: **3.4 -- 03**

RAI STATEMENT:

ITS 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits 3.--CTS 3.H.1.b requires the maximum indicated Tavg be 571.5 F. ITS 3.4.1 requires RCS average loop temperature be 571.5 F. ITS 3.4.1 Bases indicates the RCS average loop temperature is determined by calculating the average temperature for each loop and then calculating the average of these loop temperatures. This average of the averages is compared to the acceptance criteria. CTS is using the maximum indicated Tavg for the LCO but ITS is using the average Tavg. DOC A.7 states that this is an administrative change however this is considered a less restrictive condition. Provide justification for a less restrictive change.

NYPA RESPONSE:

ITS 3.4.1, Bases for Background and Bases for SR 3.4.1.2, was revised to read as follows:

RCS average loop temperature is assumed to be the highest indicated value of the Tavg indicators and this is the value that is compared to the acceptance criteria.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.2 RCS Minimum Temperature for Criticality**

NRC RAI No: **3.4 -- 04**

RAI STATEMENT:

ITS 3.4.2 Required Action A contains a typographical error misspelling "keff" as "kef".

NYPA RESPONSE:

Corrected.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.3 RCS Pressure and Temperature (P/T) Limits**

NRC RAI No: **3.4 -- 05**

RAI STATEMENT:

LCO 3.4.3, SR 3.4.3.1, PA.1 - Since IP-3 does not have a PTLR, delete reference to PTLR. Include and list the Figures that apply.

NYPA RESPONSE:

NYPA has revised the ITS conversion submittal to eliminate relocation of required information to the PTLR.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.3 RCS Pressure and Temperature (P/T) Limits**

NRC RAI No: **3.4 -- 06**

RAI STATEMENT:

CTS 3.1.B identified that Figure 3.1-1 and 3.1-2 are effective for the service period up to 13.3 effective full-power years (EFPYs). DOC LA.1 item "d" states that Figures 3.1-1 and 3.1-2 are valid up to 11.00 effective full-power years. The CTS and DOC LA.1 conflict on what the service period CTS Figures 3.1-1 and 3.1-2 are effective. Revise the DOC to correctly identify the effective period for CTS Figures 3.1-1 and 3.1-2.

NYPA RESPONSE:

NYPA has revised the ITS conversion submittal to eliminate relocation of required information to the PTLR. Therefore, the notation that Figure 3.1-1 and 3.1-2 are effective for the service period up to 13.3 will be retained in the ITS as part of Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3. This change will correct the conflict described in this RAI.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.3 RCS Pressure and Temperature (P/T) Limits**

NRC RAI No: **3.4 -- 07**

RAI STATEMENT:

CTS 3.4.3.B.1 specifies heatup and cooldown rates are averaged over one hour. Both DOC LA.1 item f. and DOC A.5 justify removal of this information from CTS. Identify the correct classification for the change and supply applicable documentation.

NYPA RESPONSE:

LA.1 was revised to eliminate relocation of this information to the PTLR consistent with the response to RAI 3.4-05.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.3 RCS Pressure and Temperature (P/T) Limits**

NRC RAI No: **3.4 -- 08**

RAI STATEMENT:

DOC LA.1 does not discuss the second sentence of CTS 3.1.B.1.a that allows interpolation of the limit lines for cooldown rates. This information is required in the DOC to justify information removal from the CTS. Provide documentation to support removal from CTS.

NYPA RESPONSE:

NYPA has revised the ITS 3.4.3, DOC LA.1, to show that the CTS 3.4.3.B.1.a stipulation that allows interpolation of the limit lines for cooldown rates is included in the information that is being relocated to the Bases. (Note: LA.1 was revised to maintain some items in the ITS and relocate others to the Bases consistent with the response to RAI 3.4-05.)

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.3 RCS Pressure and Temperature (P/T) Limits**

NRC RAI No: **3.4 -- 09**

RAI STATEMENT:

CTS 4.3.A.c states that Figure 4.3-1 is applicable for the first 13.3 EFPYs of operations. This information is not included in the list of details removed per DOC LA.1. Provide documentation for the removal of this information from CTS.

NYPA RESPONSE:

NYPA has revised the ITS conversion submittal to eliminate relocation of required information to the PTLR. Therefore, the notation that CTS 4.3.A.c Figure 4.3-1 are effective for the service period up to 13.3 will be retained in the ITS in Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3. This change will correct the conflict described in this RAI.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.3 RCS Pressure and Temperature (P/T) Limits**

NRC RAI No: **3.4 -- 10**

RAI STATEMENT:

STS 3.4.3 Required Action B.2 requires the reactor be in MODE 5 with RCS pressure < [500] psig. ITS 3.4.3 Required Action B.2 requires the reactor be in MODE 5 thereby deleting "with RCS pressure < [500] psig." JFD X.1 identifies this change as a preference by IP3. While it is true that the pressure in Mode 5 will be < [500] psig, this is a generic change and should be approved by the TSTF. Retain the STS.

NYPA RESPONSE:

NYPA has revised the ITS submittal to retain STS 3.4.3, Required Action B.2, which requires the reactor be in "MODE 5 with RCS pressure < 500 psig" if RCS pressure, temperature, heatup, or cooldown limits are not met.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.4 RCS Loops - MODES 1 and 2**

NRC RAI No: **3.4 -- 11**

RAI STATEMENT:

CTS 2.C (3), page 3 of 6 (There are no other identifiers, so it is unclear what topic the material is under), DOC A.3 deletes a licensing restriction that prohibits operating at power levels above levels defined in the FFD and FSAR and identifies it as an administrative change. CTS license condition 2.c. (3) requires 4 RCPs with power levels above 10% rated power. This change should be reviewed by Tech Staff to ensure that the license restriction is no longer needed. This is a Beyond Scope issue.

NYPA RESPONSE:

ITS 3.4.4, DOC A.3 deletes License Condition which states:

Facility Operating License DPR-64, paragraph 2.C (3), Less Than Four Loop Operation, specifies that the reactor shall not be operated at power levels above P-7 (as defined in Section 7.2 of the Final Facility Description and Safety Analysis Report (i.e., 10% RTP)) with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensee and approval for less than four loop operation at power levels above P-7 has been granted by the Commission and amendment of this license.

This License Condition 2.C (3) was originally intended to prevent 3 loop operation and can be deleted as an administrative change because either of the following ITS LCOs will prevent any power operation with less than 4 RCPs in operation:

ITS LCO 3.4.1 requires that RCS total flow rate is greater than 375,600 gpm at all times in MODE 1: and/or

ITS LCO 3.4.4 requires that 4 RCPs are Operable and in operation at all times in MODES 1 and 2.

Therefore, License Condition 2.C (3) is redundant and unnecessary and should be deleted because it is less restrictive than at least 2 LCOs and, therefore, potentially confusing.

ITS LCO: **3.4.4 RCS Loops - MODES 1 and 2**

NRC RAI No: **3.4 -- 12**

RAI STATEMENT:

Changes referenced by DOCs A.4 and A.5 do not appear to be administrative. Are they less or more restrictive?

NYPA RESPONSE:

ITS 3.4.4, DOCs A.4 and A.5, have been reclassified as more restrictive changes M.3 and M.4, respectively.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.4 RCS Loops - MODES 1 and 2**

NRC RAI No: **3.4.4 -- 00**

RAI STATEMENT:

None

NYPA RESPONSE:

None required.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.5 RCS Loops - MODE 3**

NRC RAI No: **3.4 -- 13**

RAI STATEMENT:

CTS 3.1.A.1.b.2 prohibits control bank withdrawal unless four reactor coolant pumps are operating. ITS 3.4.5 requires two RCS loops be in operation when the Rod Control System is capable of rod withdrawal. DOC L.1 justifies this change because analysis for Vantage 5 fuel only requires two RCPs in operation during a startup rod withdrawal accident. The restriction on control bank withdrawal with less than 4 RCPs when the reactor is subcritical with RCS Tavg > 350F is necessary to conform to assumptions used in transient analysis for uncontrolled control rod withdrawal event from subcritical condition. FSAR safety analysis assumes all 4 RCPs to be operating within the temperature range of concern as stated in Basis, page 3.1-7. This statement does not say two or more. Is this an unanalyzed event. Provide further discussion and justification for this change. The DOC also does not include a discussion of the controls that will assure that Technical Specifications will be evaluated if a future change in fuel type occurs.

NYPA RESPONSE:

FSAR 14.1-14, Rev 5, dated 12/97, specifies that only two RCPs are assumed to be in operation during a startup rod withdrawal accident. Research indicates that this assumption goes back to governing analyses as early as 1988. It appears that the CTS requirement (i.e., 4 RCPs in operation) and associated CTS Bases were never updated to reflect the standard Westinghouse analysis assumption that 2 RCPs are in operation during a startup rod withdrawal accident.

10 CFR 50.59 will ensure that Technical Specifications will be evaluated if a future change in fuel type occurs.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.5 RCS Loops - MODE 3**

NRC RAI No: **3.4 -- 14**

RAI STATEMENT:

In LCO 3.4.5 NOTE b., PA.1 changes "at least" to " ." Both the CTS and STS contain "at least." Retain STS

NYPA RESPONSE:

Corrected.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.5 RCS Loops - MODE 3**

NRC RAI No: **3.4 -- 15**

RAI STATEMENT:

STS 3.4.5 LCO is modified in the ITS to incorporate Revision 0 of TSTF-153. This TSTF only changes "be de-energized" to "not be in operation." No other changes are made by TSTF 153. Therefore, Insert: B3.4-22-01 and the deletion at page B 3.4-23 justified by T.1 are not acceptable. Retain STS.

NYPA RESPONSE:

Insert: B3.4-22-01 replaces two paragraphs of a very detailed description of tests that are performed during initial startup testing with a more general statement that the allowance applies to tests and maintenance that can only be performed with RCPs secured. This insert was incorrectly labeled as being part of TSTF-153.

NYPA revised the ITS to correctly identify Insert: B3.4-22-01 as JD PA.1. PA.1 applies to a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. This change made no technical changes to requirements as specified in NUREG 1431, Rev. 1, except to delete material that applies only to plants undergoing initial startup testing.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.5 RCS Loops - MODE 3**

NRC RAI No: **3.4 -- 16**

RAI STATEMENT:

ITS JFD DB.1 and DB.2 provide justifications that are not clear as to whether they are contained in your CTS or not. The numerous changes justified by these JFDs should be justified separately where they differ from your design basis. Insert: B 3.4-21-01 mentions that calculations have shown that the reactor decay heat.... It also references an analysis (Ref. 1). What is Ref. 1 and should these or have these analyses been approved by the NRC.

NYPA RESPONSE:

This statement is verbatim from FSAR 14.1.6, page 14.1-59, Rev 7/90. The reference was left out of the summary list by mistake.

NYPA revised the ITS conversion submittal to include FSAR 14.1.6 as a reference.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.5 RCS Loops - MODE 3**

NRC RAI No: **3.4 -- 17**

RAI STATEMENT:

ITS SR 3.4.5.2 change is justified by DB.2 for the 71% (wide range) change. Is this your licensing basis and is it in keeping with your analysis assumptions.

NYPA RESPONSE:

The IP3 CTS, FSAR and SER do not specify any required steam generator water level for crediting the SG as the backup decay heat removal method when in MODES 3, 4 or 5. IP3 is voluntarily adopting ITS SR 3.4.5.2 which will require verification of SG OPERABILITY if two SGs are being credited as the backup decay heat removal method. IP3 selected the acceptance criteria for this SR as a SG level that will ensure that SG tubes are covered which is identical to the acceptance criteria specified in NUREG-1431.

NYPA revised the acceptance criteria for SR 3.4.5.2 to 71% wide range (i.e., deleted the term equivalent).

NYPA included the following revised clarification in the Bases to improve clarity and ensure requirements are fully understood and consistently applied:

SG OPERABILITY is verified by ensuring that the secondary side water level is greater than or equal to 71% wide range for each required loop. Depending on plant conditions, either wide range or narrow range SG level instruments may be used to verify this SR is met. Operators may be required to adjust the indicated level to compensate for the effects of SG temperature.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.6 RCS Loops - MODE 4**

NRC RAI No: **3.4 -- 18**

RAI STATEMENT:

All of the comments from ITS 3.4.5 that apply here should also be addressed in all of the following specifications where they apply.

NYPA RESPONSE:

NYPA evaluated each of the ITS LCO 3.4.5 RAIs for applicability to ITS LCO 3.4.6. As a result of this review, the following responses are provided:

RAI 3.4-13, pertaining to number of RCPs in operation as an initial condition for a uncontrolled rod withdrawal from the source range, is not applicable to ITS LCO 3.4.6.

RAI 3.4-14, pertaining to use of the term "at least" versus an equality symbol, was incorporated into ITS LCO 3.4.6.

RAI 3.4-15 is applicable to ITS LCO 3.4.6. NYPA revised the ITS conversion submittal to correctly identify Insert: B3.4-28-01 as JD PA.1.

RAI 3.4-16 is not applicable to ITS LCO 3.4.6.

RAI 3.4-17 is applicable to ITS LCO 3.4.7 and the response to RAI 3.4-17 is also applicable to ITS LCO 3.4.7.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.6 RCS Loops - MODE 4**

NRC RAI No: **3.4 -- 19**

RAI STATEMENT:

DOC M.2 discusses the addition of ITS SR 3.4.6.1, 3.4.6.2, and 3.4.6.3 to the ITS. DOC M.2 is not identified on the CTS markup. Show addition of SRs on CTS.

NYPA RESPONSE:

NYPA revised the ITS conversion submittal to include a CTS markup notation to DOC M.2, the discussion of the addition of ITS SR 3.4.6.1, 3.4.6.2, and 3.4.6.3 to the ITS.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.6 RCS Loops - MODE 4**

NRC RAI No: **3.4 -- 20**

RAI STATEMENT:

STS 3.4.6 NOTE 2 and DB.3 prevents starting reactor coolant pumps with one or more RCS cold leg temperatures [275] F unless the secondary side water temperature of each SG is [50] F above each of the RCS cold leg temperature. ITS 3.4.7 NOTE 2 prohibits starting reactor coolant pumps unless the average the RCS cold leg temperatures less than the LTOP enable temperature unless the requirements of LCO 3.4.12, Low Temperature Overpressure Protection, are met. IP3 identified this as a plant specific difference in the design or design basis, however, no information was provided that verifies that this change is required by a plant specific design difference. The STS requires this heat addition analysis and the proposed substitution is less restrictive because it will allow and RCP startup with SG temperature more than 50 F higher than the RCS temperature. LCO 3.4.12 does not include the restriction of the maximum temperature difference of 50 F between the RCS and the SG secondary water. This condition is beyond the LTOP analysis assumptions. Retain the ITS.

NYPA RESPONSE:

STS 3.4.6, NOTE 2, is intended to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. The 50 F limit on temperature difference between an RCS loop and SG secondary side is a standard assumption for the LTOP analysis of an RCP start.

IP3 analysis assumptions for an RCP start under LTOP conditions are included in CTS 3.1.A.1.h.1, 3.1.A.1.h.2 and 3.1.A.1.h.3. These requirements include 6 separate parameters when SG temperature is less than the coldest RCS loop temperature and 5 different parameters if SG temperature is greater than RCS loop temperature. The RCP pump start when in LTOP requirements have been maintained as ITS SR 3.4.12.8 and SR 3.4.12.9.

Maintaining these requirements in ITS SR 3.4.12.8 and SR 3.4.12.9 instead of as a Note to LCO 3.4.6 is necessitated by the complexity of the IP3 requirements. Therefore, IP3 will maintain the ITS LCO 3.4.6, Note 2, as written. This is consistent with CTS Amendment 179, dated April 1998.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.6 RCS Loops - MODE 4**

NRC RAI No: **3.4 -- 21**

RAI STATEMENT:

Insert: B.4-31-01, DB.2 is confusing in that words are added that state "...either wide range or narrow range SG level instruments may be used.... Since the wide range and narrow range measurements are different, i.e., narrow range at perhaps somewhere near 17% and the wide range will be about 72 %, how can the statement be true. Delete this information, retain the ITS.

NYPA RESPONSE:

See Response to RAI 3.4-17.

NYPA revised the acceptence criteria for SR 3.4.6.2 to 71% wide range (i.e., deleted the term equivalent).

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.7 RCS Loops - MODE 5, Loops Filled**

NRC RAI No: **3.4 -- 22**

RAI STATEMENT:

ITS 3.4.7, RCS Loops – MODE 5, Loops Filled

See all applicable previous comments.

NYPA RESPONSE:

NYPA evaluated each of the ITS LCO 3.4.5 and 3.4.6 RAIs for applicability to ITS LCO 3.4.7. As a result of this review, the following responses are provided:

RAI 3.4-13, pertaining to number of RCPs in operation as an initial condition for a uncontrolled rod withdrawal from the source range, is not applicable to ITS LCO 3.4.7.

RAI 3.4-14, pertaining to use of the term "at least" versus an equality symbol, was incorporated into ITS LCO 3.4.7.

RAI 3.4-15 is applicable to ITS LCO 3.4.7. NYPA revised the ITS conversion submittal to correctly identify Insert: B3.4-33-03 as JD PA.1.

RAI 3.4-16 is not applicable to ITS LCO 3.4.7.

RAI 3.4-17 is applicable to ITS LCO 3.4.6 and the response to RAI 3.4-17 is also applicable to ITS LCO 3.4.7.

RAI 3.4-18 is applicable to ITS LCO 3.4.7 and is addressed by the five line items above.

RAI 3.4-19 is not applicable to ITS LCO 3.4.7.

RAI 3.4-20 is applicable to ITS LCO 3.4.7 and the response to RAI 3.4-20 is also applicable to ITS LCO 3.4.7.

RAI 3.4-21 is applicable to ITS LCO 3.4.7 and the response to RAI 3.4-21 is also applicable to ITS LCO 3.4.7.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.7 RCS Loops - MODE 5, Loops Filled**

NRC RAI No: **3.4 -- 23**

RAI STATEMENT:

CTS 3.1.7.a requires that if less than two RHR systems are OPERABLE initiate corrective action to return required equipment to an OPERABLE status as soon as possible. The corresponding ITS 3.4.7, Action A, requires one RHR loop inoperable AND required SGs secondary side actual water level be outside of the allowed limits before immediate action is required to restore a second RHR loop to OPERABLE status. This less restrictive change is not discussed or justified in the submittal. Provide applicable change documentation.

NYPA RESPONSE:

NUREG-1431 for ITS 3.4.7 shows that DOC L.1 addresses both the new option of allowing use of two SGs as a backup decay heat removal method and the actions required if this option is selected but requirements are not met. Minor editorial comments made to DOC L.1 were made to improve clarity.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.7 RCS Loops - MODE 5, Loops Filled**

NRC RAI No: **3.4 -- 24**

RAI STATEMENT:

STS 3.4.7.b requires secondary side water level of at least two steam generators be above a specific level. PA.1 or DB.1 (it is not clear which) justifies ITS 3.4.7.b deviation from the STS by only requiring one steam generator water level be greater than a specific level. Reducing the number of steam generators capable of heat removal was not justified and it is not readily apparent that this is in your CTS. Provide roadmap in the CTS for the STS deviation requiring only one steam generator level to be greater than the required level. Otherwise, retain ITS.

NYPA RESPONSE:

NYPA revised the IP3 ITS conversion submittal to required at least two Operable SGs when using SGs as the backup decay heat removal system.

INSERT: B 3.4-32-03 was added by NYPA to ensure that the guidance from NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation" was available to the operators. Additionally, INSERT: B 3.4-34-03 was added by NYPA to ensure that the conditions necessary to support natural circulation are maintained.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.8 RCS Loops - MODE 5, Loops Not Filled**

NRC RAI No: **3.4 -- 25**

RAI STATEMENT:

ITS 3.4.8, RCS Loops – MODE 5, Loops Not Filled
See all applicable previous comments.

NYPA RESPONSE:

NYPA evaluated each of the ITS LCO 3.4.5, 3.4.6 and 3.4.7 RAIs for applicability to ITS LCO 3.4.8. As a result of this review, the following responses are provided:

RAI 3.4-13, pertaining to number of RCPs in operation as an initial condition for a uncontrolled rod withdrawal from the source range, is not applicable to ITS LCO 3.4.8.

RAI 3.4-14, pertaining to use of the term at least" versus an equality symbol, was incorporated into ITS LCO 3.4.8.

RAI 3.4-15 is not applicable to ITS LCO 3.4.8.

RAI 3.4-16 is not applicable to ITS LCO 3.4.8.

RAI 3.4-17 is not applicable to ITS LCO 3.4.8.

RAI 3.4-18 is applicable to ITS LCO 3.4.8 and is addressed by the five line items above.

RAI 3.4-19 is not applicable to ITS LCO 3.4.8.

RAI 3.4-20 is not applicable to ITS LCO 3.4.8.

RAI 3.4-21 is not applicable to ITS LCO 3.4.8.

RAI 3.4-22 is applicable to ITS LCO 3.4.8 and is addressed by the line items above..

RAI 3.4-23 is not applicable to ITS LCO 3.4.8.

RAI 3.4-24 is not applicable to ITS LCO 3.4.8.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.9 Pressurizer**

NRC RAI No: **3.4 -- 26**

RAI STATEMENT:

CTS 3.1.C.4, DOC L.1 and LCO 3.4.9 Applicability identify a pressurizer water level at a bracketed [92%]. This number, based on Westinghouse analysis is closer to 60%. CTS 3.1.A.1.h(3) has the level at 73%. The number to be used is the one assumed in your analysis.

NYPA RESPONSE:

IP3 analysis for overpressure events identifies 58.3% as the analytical initial condition for pressurizer level. NYPA revised the ITS conversion submittal to identify 58.3% as the LCO for pressurizer limit with an explanation in the Bases that a margin of 7% is allowed for instrument error.

CTS 3.1.A.1.h(3) is an LTOP limit for an initial condition for mass injection and is not related to pressurizer level during normal operation. The CTS 3.1.A.1.h(3) pressurizer level limit is maintained in LCO 3.4.12, LTOP.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.9 Pressurizer**

NRC RAI No: **3.4 -- 27**

RAI STATEMENT:

ITS Applicability 3.4.9 b. and DB.1 deletes the ITS reference to "and capable of being powered from an emergency power supply." The CTS Basis, page 3.1-8, DOC A.1 which has been deleted shows the same requirement. Retain ITS Applicability and SR 3.4.9.3. Otherwise, if plant specific, identify as a Beyond Scope issue.

NYPA RESPONSE:

STS LCO 3.4.9.b states requires: Two groups of pressurizer heaters OPERABLE with the capacity of each group [125] kW [and capable of being powered from an emergency power supply].

NYPA revised the ITS to maintain the requirement as stated in NUREG-1431 with the clarification in the bases that each group should be powered from a separate power supply.

However, this LCO statement was written for plants with "dedicated heaters" and is designed to work in conjunction with STS SR 3.4.9.3 (Verify each group of heaters can be powered from an emergency power supply.)

Because the IP3 design is not consistent with the design of the STS LCO, IP3 investigated the source of the requirement which is as follows:

NUREG-0737, Section II.E.3., EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS, which states: Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17, and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be Implemented:
(1) The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

As stated in the Bases, IP3 has multiple groups of pressurizer heaters that can be used to satisfy requirements. Therefore, bracketed sections of the STS were revised as necessary to make the IP3 ITS satisfy the intent of NUREG-0737, Section II.E.3.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.9 Pressurizer**

NRC RAI No: **3.4 -- 28**

RAI STATEMENT:

ITS 3.4.8.2, T.1 indicates that TSTF 93 allows a 24 month frequency. Actually, TSTF provides for a frequency of 18 months for non-dedicated heaters. If you have non-dedicated heaters, change the frequency to 18 months, Otherwise retain 92 days.

NYPA RESPONSE:

IP3 uses normal (i.e., non-dedicated) heaters to satisfy requirements.

TSTF-093, Rev 3, identifies the required Frequency for testing non-dedicated heaters as [18] months with the [ ] indicating that the Frequency can be adjusted to satisfy plant specific requirements.

IP3 voluntarily elected to perform this SR 3.4.8.2 (ITS 3.4.9, DOC M.2) at the 24 month Frequency. The 24 month Frequency was selected (as explained in the Bases and DOC M.4) because the SR is satisfied by performing an electrical check on heater element continuity and resistance (i.e., it must be performed when the rector is shutdown). The Frequency of 24 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

Therefore, IP3 will maintain the 24 month Frequency for ITS SR 3.4.8.2.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.10 Pressurizer Safety Valves**

NRC RAI No: **3.4 -- 29**

RAI STATEMENT:

CTS 3.2.b requires all pressurizer code safety valves to be Operable. ITS 3.4.10 has been changed to three. Does IP-3 only have three safety valves. Provide discussion and justification for difference.

NYPA RESPONSE:

As stated in FSAR 8.2, Two power operated relief valves (PORVs) and three code safety valves (SVs) are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray. CTS Bases (page 3.1-7) further states: The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load (2) without a direct reactor trip or any other control. Reference (2) refers to FSAR Section 14.1.8.

NYPA revised the ITS conversion submittal to explain "All" safety valves equals 3 safety valves in DOC A.5.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.10 Pressurizer Safety Valves**

NRC RAI No: **3.4 -- 30**

RAI STATEMENT:

See comment 20. STS 3.4.10, Applicability, is MODES 1, 2, 3, and MODE 4 with all RCS cold leg temperatures 275 degrees F. ITS 3.4.10, Applicability, is MODES 1, 2, and 3, and MODE 4 with the average of the RCS temperatures greater than or equal to the LTOP arming temperature specified in LCO 3.4.12, Low Temperature Overpressure Protection.

NYPA RESPONSE:

NYPA revised the ITS conversion submittal to retain the LTOP temperature in the applicability versus a cross reference to LCO 3.4.12, LTOP.

IP3's use of the "average of the RCS cold leg temperatures" versus "all RCS cold leg temperatures" is addressed in the response to RAI 3.4-31.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.10 Pressurizer Safety Valves**

NRC RAI No: **3.4 -- 31**

RAI STATEMENT:

ITS Applicability Mode 4 has been changes from "All" to "Average." It is not clear if the JFD is DB.3 because no DB.3 justification is included. The CTS does not specify "Average." This change is generic. Retain ITS.

NYPA RESPONSE:

NYPA revised the ITS conversion submittal to adopt NUREG-1431.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 32**

RAI STATEMENT:

CTS 3.1.A.4 requires the block valve for inoperable PORVs to be closed with control power for the block valve removed. ITS 3.4.11, Condition A, requires block valve closure and maintains power to the block valve when the PORV is capable of being manually cycled. DOC M.1, item a, documented this as a more restrictive change, is maintaining power to the block valve a less restrictive than removing control power to the block valve. Provide additional discussion and justification for this change.

NYPA RESPONSE:

NYPA treats this as a more restrictive change because the CTS allows continued operation with the safety function of the PORV lines disabled (i.e., redundant manual venting capability); whereas, the ITS requires that the safety function of the PORVs (i.e., redundant manual venting capability) must be maintained if plant operation is allowed to continue. Additionally, the CTS requires that the PORV line safety function be deliberately disabled (i.e., venting capability disabled) when the pressure relieving capability safety function becomes inoperable.

ITS must allow a short period of time with leak isolation protection slightly reduced (i.e., a valve in the line with the inoperable valve must be closed but not de-energized). Another way of stating the difference is that the CTS does not recognize any safety function for the PORVs whereas the ITS recognizes that the redundant venting capability of the PORV lines is a safety function that must be maintained.

The net effect of the change is a substantial improvement to safety. Therefore, NYPA considers this a more restrictive change.

ITS 3.4.11, DOC M.1, provides a very detailed explanation of this change and why it is more restrictive.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 33**

RAI STATEMENT:

CTS 3.1.A.5 requires motor operated block valves to be OPERABLE, or closed with control power removed. The corresponding ITS 3.4.11, Condition C, requires placing the associated PORV in manual control within one hour and restoring the block valve to OPERABLE status within seven days. Closing a block valve and removing control power is more restrictive on operations than placing a PORV in manual operation. Also, closing the block valve and removing control power would provide positive protection against a leaking PORV or a PORV that fails in the open condition. Is this a more restrictive change. Provide additional discussion and justification for this change.

NYPA RESPONSE:

This RAI is addressed in the response to RAI 3.4-32 and 34.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 34**

RAI STATEMENT:

ITS Conditions B.3 and C.2 have associated completion times of 72 hours. This has been changed to 7 days. This is an extension of a Completion Time and is therefore Beyond Scope.

NYPA RESPONSE:

IP3 currently has no requirement to restore the PORV safety function to Operable and is voluntarily adopting this requirement. The NYPA decision to adopt this requirement included a 7 day AOT versus the 72 hour AOT in NUREG-1431. The justification is provided in IP3 ITS 3.4.11, JD .1, which states:

IP3 ITS differs from NUREG-1431 by extending the allowable out of service time from 72 hours to 7 days for one PORV inoperable and not capable of being manually cycled (loss of redundancy of the manual venting function) (Condition B) and for one block valve inoperable (i.e., not capable of being manually cycled) (Condition C). This change is acceptable because the 7-day AOT is for loss of redundancy, not loss of function, of the manual venting that is used to reduce RCS pressure following a SGTR. During the 7-day AOT, one PORV vent path is still available for venting and both normal and alternate pressurizer spray are typically available to perform the same function.

Note that CTS never requires that manual venting capability be restored and, in fact, deliberately disables the manual venting capability when the PORV pressure relieving capability is lost.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 35**

RAI STATEMENT:

ITS SR 3.4.11.2 Frequency has been changed from 18 to 24 months. This is an extension of a Frequency and is therefore Beyond Scope.

NYPA RESPONSE:

STS 3.4.11.2 Frequency of 18 months is bracketed and CTS Table 4.1-3, Item 15, is already 24 months. Note that this SR must be performed with the reactor shutdown.

Therefore, IP3 will maintain the 24 month Frequency for ITS SR 3.4.11.2 based on CLB.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 36**

RAI STATEMENT:

CTS Table 4.1-3, item 15, describes the check of the PORVs as verifying OPERABILITY. The corresponding ITS SR 3.4.11.2 requires performing a complete cycle of each PORV. Documentation was not provided to verify that the check made by CTS Table 4.1-3, item 15, and ITS SR 3.4.11.2 are equivalent. Provide applicable documentation for the change.

NYPA RESPONSE:

CTS does not define any criteria for operability other than that the valve can be closed. STS 3.4.11 does not define any criteria for operability other than that the valve can be opened and closed. Currently, CTS Table 4.1-3, item 15, is satisfied by Procedure 3PT-CS-28 which requires only that the valves are cycled and that valve opening and closing time is within specified limits that are consistent with the valve design.

NYPA believes that CTS Table 4.1-3, item 15, requirements for Operability are identical to those described in STS SR 3.4.11.2.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 37**

RAI STATEMENT:

STS SR 3.4.11.3 and 3.4.11.4 are deleted from the ITS. There is no discussion or justification for the STS deviation. Retain ITS.

NYPA RESPONSE:

SR 3.4.11.3 is bracketed and is required for plants with pilot operated PORVs. IP3's PORVs are not pilot operated; therefore, SR 3.4.11.3 does not apply to IP3. Additionally, IP3 accumulators are sized to provide sufficient energy for the PORV to perform its design function and IP3 does not have individual solenoid air control valves on the accumulators to provide makeup to the accumulators.

SR 3.4.11.4 is bracketed and does not apply to plants with permanent vital power supplies to the PORVs and block valves. As indicated in the Background section of the Bases, these valves are powered from vital buses. Therefore, this SR does not apply to IP3.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 38**

RAI STATEMENT:

STS 3.4.11 Required Action F.3 is deleted from the ITS. There is no discussion or justification for the STS deviation. Retain ITS.

NYPA RESPONSE:

STS 3.4.11, Required Action F.3, is bracketed and like the other bracketed portions of the Required Actions applies only to plants with more than 2 PORVs. Note the other text deleted in the STS Required Action F.2 and the explanation in the Bases.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 39**

RAI STATEMENT:

ITS Bases 3.4.11 - there are a number of Bases changes that have no justification and can not be found in the CTS. Identify the plant specific changes or retain CTS.

NYPA RESPONSE:

NYPA reviewed each of the changes to the LCO 3.4.11 Bases and confirmed that each of the changes is needed either to describe the IP3 design or to provide a more detailed discussion of an existing requirement in the Bases. NYPA revised the ITS conversion submittal to provide the required classification for each of these changes.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.11 Pressurizer Power Operated Relief Valves (PORVs)**

NRC RAI No: **3.4 -- 40**

RAI STATEMENT:

Some ITS Bases changes appear to be attributed to TSTF 151 that are not reflected in the approved version of TSTF 151. Delete those changes justified by TSTF 151 that are not a part of TSTF 151.

NYPA RESPONSE:

NYPA revised the submittal and incorporate TSTF 151, Revision 1. Revision 0 of the ITS conversion submittal was based on Revision 0 of TSTF 151.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.12 Low Temperature Overpressure Protection (LTOP)**

NRC RAI No: **3.4 -- 41**

RAI STATEMENT:

There are so many changes with additions and deletions, and information from specifications other than the Overpressure Protection Specification with some not justified that this complete specification should be reviewed as a Beyond Scope.

NYPA RESPONSE:

NYPA believes that the ITS conversion to IP3 ITS 3.4.12, Low Temperature Overpressure Protection, appropriately adopts STS 3.4.12 while maintaining current licensing basis as approved in CTS Amendment 179, dated 4 April 98.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.12 Low Temperature Overpressure Protection (LTOP)**

NRC RAI No: **3.4 -- 42**

RAI STATEMENT:

Reference to the PTLR should be deleted since IP-3 does not have a PTLR.

NYPA RESPONSE:

NYPA revised the ITS conversion submittal to eliminate relocation of any requirements to the PTLR.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.13 RCS Operational LEAKAGE**

NRC RAI No: **3.4 -- 43**

RAI STATEMENT:

CTS Table 4.1-3, Item 7, lists the check for Primary System Leakage as "Evaluate". DOC A.5 is referenced for the change but does not discuss this change. DOC A.5 discusses the addition of ITS SR 3.4.13.2 to verify that Steam Generator Tube Integrity is in accordance with the Steam Generator Tube Surveillance Program. DOC A.5 does not discuss primary system leakage other than Steam Generator Tube Integrity. This appears that this change should reference DOC A.7. Provide applicable documentation for the change.

NYPA RESPONSE:

CTS markup of Table 4.1-3, Item 7, incorrectly references DOC A.5. The correct reference is DOC A.7.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.13 RCS Operational LEAKAGE**

NRC RAI No: **3.4 -- 44**

RAI STATEMENT:

Is it correct to assume that the Bases changes that do not have a JFD are your licensing basis.

NYPA RESPONSE:

NYPA reviewed each of the changes to the LCO 3.4.13 Bases and confirmed that each of the changes is needed either to describe the IP3 design or to provide a more detailed discussion of an existing requirement in the Bases. The majority of the changes are to accommodate IP3s allowance of excluding leakage into closed systems from identified and unidentified leakage limits. NYPA revised the ITS conversion submittal to identify the appropriate JFD.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.14 RCS Pressure Isolation Valve (PIV) Leakage**

NRC RAI No: **3.4 -- 45**

RAI STATEMENT:

STS SR 3.4.14.1, Frequency, of "In accordance with the Inservice Testing Program," was omitted in ITS SR 3.4.14.1. No discussion or justification was provided from omitting this requirement. Provide applicable documentation for the deviation from the STS

NYPA RESPONSE:

STS SR 3.4.14.1 lists the Frequency as:
"In accordance with the Inservice Testing Program, and [18] months."

This dual SR Frequency requirement is unique in NUREG-1431 and is both ambiguous and confusing. Therefore, IP3 elected to maintain the CTS 4.5.B.2.c which is 24 months. However, IP3 will maintain the NUREG-1431 Bases for this SR Frequency which state:

"Testing is to be performed every 24 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.14 RCS Pressure Isolation Valve (PIV) Leakage**

NRC RAI No: **3.4 -- 46**

RAI STATEMENT:

STS SR 3.4.14.2 and 3.4.14.3, Note, was omitted in ITS 3.4.14.2 and ITS SR 3.4.13.3. The omitted Note refers to STS SR 3.4.12.7 which was also omitted in the ITS. Refer to comment #48 concerning STS SR 3.4.12.7.

NYPA RESPONSE:

STS SR 3.4.14.2 and 3.4.14.3, which test the RHR suction auto closure interlock, includes a bracketed Note which states: "Not required to be met when the RHR System auto closure interlock is disabled in accordance with SR 3.4.12.7." This Note is needed in the STS because STS 3.4.12 provides an option that uses the RHR relief valves to be used as the LTOP relief valve(s). This Note cannot be included in the IP3 ITS because RHR relief valves can never be used to provide LTOP at IP3. The RHR relief valves at IP3 are sized to the capacity of 3 charging pumps and do not provide adequate vessel protection. In fact, IP3 requires that LTOP requirements be met whenever the RHR system is not isolated from the RCS even when above LTOP temperatures because LTOP requirements are used to protect the RHR system from over pressurization.

NYPA revised ITS to include the explanation above as JFD DB.2.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.14 RCS Pressure Isolation Valve (PIV) Leakage**

NRC RAI No: **3.4 -- 47**

RAI STATEMENT:

ITS LCO 3.4.14, Applicability, Mode 4, PA.2 adds clarifying words. This change, while acceptable, is generic and should be addressed through the TSTF.

NYPA RESPONSE:

See Response to RAI 3.4-51.

Without this change, requirements for the RHR auto closure interlock will not apply in Mode 4 which is contrary to the design intent for the auto closure interlock as explained in WCAP-11736-A.

NYPA plans to prepare and submit a TSTF to correct NUREG-1431, Required Action C.1 and ITS LCO 3.4.14, Applicability, so that it is consistent with WCAP-11736-A. The ITS conversion submittal is written based on the WCAP.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.14 RCS Pressure Isolation Valve (PIV) Leakage**

NRC RAI No: **3.4 -- 48**

RAI STATEMENT:

ITS LCO 3.4.14, Condition C.1 has an added Note that is justified by X.1. This change is Beyond Scope since it is not included in the CTS or the STS.

NYPA RESPONSE:

See Response to RAI 3.4-51.

The added note is consistent with the IP3 CLB and WCAP-11736-A. NYPA plans to submit a TSTF change request to propose revised wording for NUREG-1431.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.14 RCS Pressure Isolation Valve (PIV) Leakage**

NRC RAI No: **3.4 -- 49**

RAI STATEMENT:

SR 3.4.14.1, SR 3.4.14.2 and SR 3.4.14.3 Frequencies are all extended from 18 to 24 months. These are extensions of Frequencies and are Beyond Scope.

NYPA RESPONSE:

STS SR 3.4.14.1 has a Frequency that is bracketed and IP3 SR 3.4.14.1 maintains requirements in CTS 4.5.B.2.c which already has a Frequency of 24 months.

STS SR 3.4.14.2 and SR 3.4.14.3 have Frequencies that are bracketed and IP3 SR 3.4.14.2 and SR 3.4.14.3 maintain requirements in CTS Table 4.1-3, Item 13, which already has a Frequency of 24 months.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.14 RCS Pressure Isolation Valve (PIV) Leakage**

NRC RAI No: **3.4 -- 50**

RAI STATEMENT:

SR 3.4.14.1, Frequency also includes another extension of the Frequency for testing prior to entering Mode 2 from 8 to 12 months. This is Beyond Scope.

NYPA RESPONSE:

IP3 SR 3.4.14.1 and the supporting Bases differ from NUREG-1431, Rev 1, in that the limit on the conditional Frequency was changed from 9 months to 12 months. The conditional Frequency is intended to approximate the mid point in a normal refueling cycle. Therefore, the IP3 normal Frequency of 24 months for SR 3.4.14.1 supports a conditional Frequency of 12 months.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.14 RCS Pressure Isolation Valve (PIV) Leakage**

NRC RAI No: **3.4 -- 51**

RAI STATEMENT:

Is Insert B 3.4-80-01 your licensing basis.

NYPA RESPONSE:

Yes. See FSAR Section 6.2.

This insert is also consistent with WCAP-11736-A, "Residual Heat Removal System Auto closure Interlock (ACI) Removal Report," which provides the definitive license and design basis for the RHR auto closure (ACI) and open permissive (OPI) interlocks for all Westinghouse plants.

Based on a detailed review of WCAP-11736-A performed after the IP3 ITS submittal, NUREG-1431, Required Action C.1, and its associated Bases appear to be misleading and/or incorrect. In particular, the second sentence of the following excerpt of the Bases for Required Action C.1 is not correct:

"If the RHR auto closure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the auto closure function."

NUREG-1431, Required Action C.1, and associated Bases appears to be based on the assumption that the interlocked RHR valves are for containment isolation when in fact the valves and interlock are for RCS boundary isolation.

WCAP-11736-A states that the purpose of these interlocks is to provide a diverse backup to administrative requirements that ensure that both 730 and 731 are closed to provide a double barrier between the RCS and the RHR System when the plant is at normal operating conditions (hot and pressurized) and not in the RHR cooling mode. The interlocks are intended to prevent a situation in which the operator closes one of the isolation valves and not the other. In this situation, a single failure of the remaining barrier has the potential to cause a LOCA in which the containment and containment safeguards radionuclide protective barriers are bypassed (i.e., a non-mitigable LOCA outside containment) after the plant has returned to normal operating conditions.

This understanding of the purpose of the interlock results in significantly different Required Action and Completion Time if the interlock on one or both of the valves becomes inoperable, especially in Mode 4.

NYPA plans to prepare and submit a TSTF to correct NUREG-1431, Required Action C.1, so that it is consistent with WCAP-11736-A as stated in the reply to RAI 3.4 - 47.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.14 RCS Pressure Isolation Valve (PIV) Leakage**

NRC RAI No: **3.4 -- 52**

RAI STATEMENT:

JFD DB.1 has been used in numerous places as justification for the differences from the ITS. It is not clear whether all of these changes are your CTS. Provide additional discussion for these changes if they are not your licensing basis.

NYPA RESPONSE:

See Response to RAI 3.4-51.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.15 RCS Leakage Detection Instrumentation**

NRC RAI No: **3.4 -- 53**

RAI STATEMENT:

ITS 3.4.15, Note, specifies that LCO 3.0.4 is not applicable to the LCO. This change was discussed in DOC A.3, which justified the change as administrative because an equivalent statement did not exist in CTS. The discussion did not specify if entry into the MODE where leak detection was required to be OPERABLE, but was not OPERABLE, was allowed by CTS. Industry usage of Technical Specifications prohibits entry into a MODE without all Technical Specification required equipment OPERABLE, unless a specific statement is included in the LCO. If CTS allowed entry into a MODE where leak detection was required to be OPERABLE, but was not OPERABLE, then this is considered an Administrative change. If CTS prohibited entry, then this change is considered less restrictive. Provide documentation that explains if CTS allowed entry into a MODE when leakage detection was required, but was not OPERABLE. If CTS did not allow entry into a condition where leak detection was required, but was not OPERABLE, then provide documentation for a Less Restrictive change.

NYPA RESPONSE:

CTS does not have any requirement similar to ITS LCO and SR 3.0.4 which are being added as part of the ITS conversion (See ITS 3.0, DOC M.1). Therefore, adding a note that states LCO 3.0.4 is not applicable to ITS 3.4.15 is an administrative change.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.15 RCS Leakage Detection Instrumentation**

NRC RAI No: **3.4 -- 54**

RAI STATEMENT:

SR 3.4.15.3, SR 3.4.15.4 and SR 3.4.15.5 all have Frequencies that are extended fro 18 to 24 months. These are Beyond Scope.

NYPA RESPONSE:

STS SR 3.4.15.4 has a Frequency that is bracketed and IP3 SR 3.4.15.4 maintains requirements in CTS Table 4.1-1, Item 15.b which already has a Frequency of 24 months.

STS SR 3.4.15.3 and SR 3.4.15.5 have Frequencies that are bracketed and the SRs must be performed while the plant is shutdown. IP3 CTS do not require these SRs. IP3 is voluntarily adopting these requirements (See ITS 3.4.15, DOC M.6) and will perform these SRs at a 24 month Frequency. As stated in DOC M.6, the 24 month Frequency is based on the need to perform this SR during a refueling outage and is consistent with the demonstrated reliability of the equipment.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.16 RCS Specific Activity**

NRC RAI No: **3.4 -- 55**

RAI STATEMENT:

ITS 3.4.16, Required Action A, specifies that LCO 3.0.4 is not applicable to the LCO. This change was discussed in DOC A.3, which justified the change as administrative because an equivalent statement did not exist in CTS. Should this change be identified as less restrictive. Provide discussion and justification for change.

NYPA RESPONSE:

CTS does not have any requirement similar to ITS LCO and SR 3.0.4 which are being added as part of the ITS conversion (See ITS 3.0, DOC M.1). Therefore, adding a note that states LCO 3.0.4 is not applicable to ITS 3.4.16 is an administrative change.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.16 RCS Specific Activity**

NRC RAI No: **3.4 -- 56**

RAI STATEMENT:

CTS Table 4.1-2 requires Radiochemical (gamma) Spectral Check at a frequency of monthly. ITS SR 3.4.16.1 requires this surveillance to be performed each 7 days, consistent with the STS. This change is discussed in DOC L.3. The change from monthly to each 7 days is more restrictive. Provide documentation for this more restrictive requirement.

NYPA RESPONSE:

CTS Table 4.1-2, Item 1, requires the detailed verification of the CTS 3.1.D.1.a requirement specific activity of the primary coolant by Radiochemical (gamma) Spectral Check only once per month because performing this verification was difficult before multi-channel analyzers were readily available. Therefore, this test was supplemented by the CTS Table 4.1-2, Item 1, requirement for a check of gross activity 5 days per week.

With more reliable fuel and readily available multi-channel analyzers allowing the detailed check to be performed much more easily, the ITS SR 3.4.16.1 requirement for verification every 7 days of the gross specific activity replaces both of the verifications required in CTS Table 4.1-2, Item 1.

NYPA revised the ITS conversion submittal to change the explanation of this change from a less restrictive change in DOC L.3 to a more restrictive change in DOC M.3.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.4.16 RCS Specific Activity**

NRC RAI No: **3.4 -- 57**

RAI STATEMENT:

CTS Table 4.1-2 item 1 requires Isotopic Analysis for I-131, I-133, and I-135. The sample and analysis frequency for this analysis have been included in SR 3.4.16.2 which sample for DOSE EQUIVALENT I-131 specific activity. The Table 4.1-2 Item 1 sample type (Isotopic Analysis for I-131, I-133, and I-135) is deleted and does not appear in ITS 3.4.16. There is no discussion and justification for this deletion. There is no discussion that Table 4.1-2 Item 1, (Isotopic Analysis for I-131, I-133, I-135), is equivalent to ITS 3.4.16 sampling requirements for DOSE EQUIVALENT I-131 specific activity or that only sampling for DOSE EQUIVALENT I-131 specific activity satisfies the requirements of CTS Table 4.1-2 item 1. Provide discussion and justification for deleting Table 4.1-2 Item 1, Isotopic Analysis for I-131, I-133, I-135, such that sampling for DOSE EQUIVALENT I-131 specific activity per ITS 3.4.16 is sufficient.

NYPA RESPONSE:

CTS 3.1.D.1.a establishes an LCO limit for Dose Equivalent Iodine-131. CTS 1.15 specifies that "DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro curie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present."

ITS SR 3.4.16.2 establishes an LCO limit for Dose Equivalent Iodine-131. ITS 1.0 specifies that "DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro curies/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present.

Therefore, CTS 3.1.D.1.a requirements are identical to ITS SR 3.4.16.2 requirements.

There is no CTS requirement for periodic verification of CTS 3.1.D.1.a, Dose Equivalent Iodine-131: however, CTS Table 4.1-2, item 1, requires Isotopic Analysis for I-131, I-133, and I-135 at the identical Frequency that ITS SR 3.4.16.2 requires verification of Dose Equivalent Iodine-131. Therefore, NYPA has always considered CTS Table 4.1-2, item 1, to be the requirement or periodic verification of CTS 3.1.D.1.a requirements consistent with the definition of Dose Equivalent Iodine-131 in CTS 1.15.

The fact that CTS Table 4.1-2, item 1, does not mention I-132 and I-134 does not relax the CTS 3.1.D.1.a requirement for Dose Equivalent Iodine-131 consistent with the CTS definition and, therefore, is either a minor administrative error in the CTS or a shorthand notation for the Dose Equivalent Iodine-131 requirement.

NYPA revised the IP3 ITS conversion submittal to include ITS 3.4.16, DOC A.7, to describe the correction of this administrative error in the CTS.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.5.2 ECCS - Operating**

NRC RAI No: **3.5.2--01**

RAI STATEMENT:

--CTS 3.3.A.4
--DOC L.2
--ITS 3.5.2, Condition A and Bases
--JFD DB.2

CTS 3.3.A.4 allows one SI pump, one RHR pump, or one RHR heat exchanger to be inoperable for a certain period of time. CTS 3.3.A.4 also allows any valve required for the functioning of the SI and RHR system to be inoperable for 24 hours. STS 3.5.2, Action A, allows one or more trains to be inoperable provided at least 100% of the ECCS flow equivalent to a single operable ECCS train is available. You have revised the STS requirement in ITS 3.5.2 to accommodate your design basis of three ECCS trains.

Comment: The staff believes that your proposed revision of STS Condition A needs clarification. The proposed ITS 3.5.2, Condition A reads, "One or more trains inoperable AND At least 100% of the ECCS flow equivalent to OPERABLE ECCS trains available". The staff believes that this statement is not definitive and should read, "One or more trains inoperable AND At least 100% of the ECCS flow equivalent to two OPERABLE ECCS trains available." The staff's suggestion is based on your proposed Bases description of the three ECCS systems which states, "The three ECCS systems (3 HHSI, 2 RHR, and 2 Recirculation) are grouped into three trains (5A, 2A/3A, and 6A) such that any 2 of the 3 trains are capable of meeting all ECCS capability assumed in the accident analysis." Please revise the proposed ITS wording of Condition A or explain why the suggested revision is not appropriate. In either case, the staff believes the wording as proposed requires some modification.

NYPA RESPONSE:

The word "two" was inadvertently left out of the second part of ITS 3.5.2, Condition A. However, this presentation is confusing; therefore, NYPA revised condition A to read as follows:

One or more trains inoperable.
AND
Two HHSI pumps, one RHR pump and one Containment Recirculation pump are Operable.

With one or more trains inoperable and any two HHSI pumps, any one RHR pump, and any one Containment Recirculation pump are OPERABLE (i.e., 100% of the ECCS capability assumed in the accident analysis available), the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 4) and is a reasonable time for repair of many ECCS components. If 100% of the ECCS capability assumed in the accident analysis is not OPERABLE, entry into LCO 3.0.3 is required.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.5.2 ECCS - Operating**

NRC RAI No: **3.5.2--02**

RAI STATEMENT:

ITS SR 3.5.2.1 and SR 3.5.2.2
--JFD CLB.1

The CTS do not require periodic verification that each valve in the ECCS flow path is in the correct position. STS SR 3.5.2.1 requires verification of proper alignment every 12 hours of any valve that would render more than one ECCS train inoperable if mispositioned. STS SR 3.5.2.2 requires verification of proper alignment of other valves that are not locked, sealed, or otherwise secured in position every 31 days.

Comment: JFD CLB.1 states that the proposed ITS SR 3.5.2.1 and SR 3.5.2.2 differ from the STS because the RWST outlet isolation valve, SI 846, is verified in its proper position every 31 days, even though closing this valve would render more than one ECCS train inoperable. JFD CLB.1 states that a 31-day Frequency is appropriate because it is a locked manual valve that is located in a locked area. However, if SI846 is a locked valve, than ITS SR 3.5.2.2 does not apply to it. ITS SR 3.5.2.2 states, "Verify each ECCS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position." The Bases for ITS SR 3.5.2.2 specifically state, "This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking , sealing, or securing." Please reconcile this discrepancy.

NYPA RESPONSE:

ITS SR 3.5.2.1 requires verification every 12 hours that the valves listed in the SR are in the listed position with power to the valve operator removed. These valves are of the type, described in IE Information Notice No. 87-01, that can disable the function of more than one ECCS train and invalidate the accident analyses. CTS already requires that these valves are de-energized in the proper position and DOC M.1 adds a new requirement for periodic verification.

NYPA revised the proposed ITS so that ITS SR 3.5.2.1 applies to the valves listed in the CTS and JFD CLB.1 has been deleted.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.5.3 ECCS - Shutdown**

NRC RAI No: **3.5.3--01**

RAI STATEMENT:

--CTS 3.3.A.1.c & d
--DOC LA.1

CTS 3.3.A.1.c requires one RHR pump and heat exchanger together with the associated piping and valves to be operable. CTS 3.3.A.1.d requires one recirculation pump together with its associated piping and valves to be operable. The CTS markup for ITS 3.5.3 indicates that DOC LA.1 applies to the relocation of the details of system operability.

Comment: DOC LA.1 for ITS 3.5.3 references CTS 3.3.A.3.e, f, and g for this change. In addition, DOC LA.1 states that ITS 3.5.3 requires operability of three ECCS trains and states that ECCS trains are defined in the ITS 3.5.2 Bases. It appears that DOC LA.1 for ITS 3.5.2 was copied for use as DOC LA.1 for ITS 3.5.3, but was not completely modified to account for the differences between the two specifications. Please revise DOC LA.1 for ITS 3.5.3 to accurately described the current requirements and the proposed changes to those requirements.

NYPA RESPONSE:

NYPA revised the proposed ITS to correct ITS 3.5.2 DOC LA.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.5.4 Refueling Water Storage Tank (RWST)**

NRC RAI No: **3.5.4--01**

RAI STATEMENT:

--No CTS requirement
--DOC M.1
--ITS 3.5.4.1
--JFD DB.2

There are no CTS requirements with regard to RWST water temperature. ITS SR 3.5.4.1 requires verification every 24 hours that RWST borated water temperature is within limits. In ITS SR 3.5.4.1, you have proposed to modify the note found in STS SR 3.5.4.1. In addition, the maximum temperature in the note is not the same as the maximum temperature in the surveillance, as it is in the STS. The markup of the STS indicates the JFD DB.2 contains the justification for these changes. However, there is no JFD DB.2. Given that the justification is not current licensing basis, it appears that this change could be generic. Please revise the ITS to adopt the STS wording or provide an appropriate justification for these deviations from the STS that addresses why they are not generic. If they are generic, the change must go through the Technical Specification Task Force for generic approval.

NYPA RESPONSE:

NYPA revised the proposed ITS and adopted the note to ITS SR 3.5.4.1 as written in NUREG-1431. This Note now reads as follows: Only required to be performed when ambient air temperature is < 35 F or > 110 F.

Note that IP3 originally submitted ITS with a minimum RWST temperature of 40F because one event listed in FSAR used 40F as an initial condition while all of the others used 35F. The current updated FSAR shows that 35F is the initial condition for the bounding event.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.5.4 Refueling Water Storage Tank (RWST)**

NRC RAI No: **3.5.4--02**

RAI STATEMENT:

--CTS 3.3.A.5
--DOC L.3

CTS 3.3.A.5 establishes the Actions required if the ECCS systems are not restored to meet CTS requirements within specified completion times. CTS 3.3.A.5.a specifies that, if the reactor is critical when requirements are not met, then the reactor shall be in hot shutdown within 4 hours and cold shutdown within the following 24 hours. Under the same conditions, ITS 3.5.4, Required Actions C.1 and C.2, require that the reactor be in Mode 3 in 6 hours and Mode 5 in 36 hours. --

Comment: The CTS markup for ITS 3.5.4 indicates that DOC L.3 applies to the change in the time to reach Mode 5 from "the following 24 hours" to "36 hours" However, there is no DOC L.3 associated with ITS 3.5.4. Please provide DOC L.3.

NYPA RESPONSE:

Changes to CTS 3.3.A.5.a should be marked as DOC L.1. NYPA has revised the proposed ITS to make this correction.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--01**

RAI STATEMENT:

--ITS SR 3.8.1.10
--STS SR 3.8.1.14 Note 2
--Bases for ITS SR 3.8.1.10, STS Bases markup page B 3.8-28

--Note 2 for STS SR 3.8.1.14 states, "This Surveillance shall not be performed in Mode 1 or 2." This Note has not been adopted in corresponding ITS SR 3.8.1.10.

Comment:--No justification has been provided to support this proposed difference. Revise the submittal to provide the appropriate justification for the proposed difference, or conform to the STS.

NYPA RESPONSE:

NYPA revised ITS SR 3.8.1.10 (STS SR 3.8.1.14) and Bases to include the STS Note that prohibits performance of the DG endurance run in Modes 1 and 2.

This item is a duplicate of RAI 3.8.1-11.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--02**

RAI STATEMENT:

--CTS 4.6.A.3
--DOC--L7

The proposed Note 3 to ITS SR 3.8.1.12 allows the SR to be conducted on all three DGs at the same time. This SR must be conducted in Modes 5 or 6 when 2 of the 3 DGs are required to be OPERABLE.

Comment: Conducting this SR on the DGs required to be OPERABLE could cause electrical system perturbations with attendant challenges to safety systems. The licensee is encouraged to revise the Note to limit the SR to one DG at a time, to be consistent with NUREG

NYPA RESPONSE:

NYPA believes that simultaneous testing of all three DGs during a LOOP/LOCA test is not only acceptable but that this method is preferred. Significant safety benefit results from discovering common failure resulting from interdependence among DGs and/or safeguard power trains during shutdown testing versus discovering these failure modes during an actual event. This test does not compromise safety because: a) the test can only be initiated when all DGs are available and there is redundant decay heat removal; b) the plant is deliberately configured to tolerate the potential for a loss of all AC power prior to initiation of the test; and, c) the plant is restricted from performing any activity that is a precursor to a shutdown event that requires AC power for mitigation. NYPA also believes that an unplanned event during the test is unlikely to result in damage to all three safeguards power trains such that at least one of the safeguards power trains could not be re-energized immediately from either one of the 3 DGs or one of the two circuits that connect safeguards power trains to the offsite circuits. Finally, simultaneous testing of all three DGs during a LOOP/LOCA test provides significant time savings during refueling outages.

The following details are presented to support NYPA's determination that simultaneous testing of all three DGs during a LOOP/LOCA test: 1) provides significant safety benefit; and, 2) is performed in a manner that does not compromise safety.

1) NYPA's position regarding the safety benefits of simultaneous DG testing are supported by Reg. Guide 1.108, Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants, Rev. 0, August 1976, which states the following in Section C.2.d: "Testing of redundant diesel generator units 'during normal plant operation' should be performed independently (nonconcurrently) to minimize common failure modes resulting from undetected interdependence among diesel generator units. However, during ... pre-operational

testing and 'once a year thereafter,-a test should be conducted where redundant units are started simultaneously to help identify certain common failure modes undetected in single diesel generator tests.'

Note the sections enclosed in single quotation marks. These statements indicate that the recommendation against simultaneous testing applies only 'during normal plant operation' and that 'once a year thereafter, a test should be conducted where redundant units are started simultaneously to help identify certain common failure modes undetected in single diesel generator tests.'

RG 1.108, Rev 2, changed the frequency for simultaneous test from every year to every 10 years (during a plant shutdown). NYPA believes that the extension to 10 years was intended to be a relaxation and not a restriction and the clarification (during a plant shutdown) was intended to be a restriction. Note also that the RG 1.9 does not include any prohibition against simultaneous DG testing in the description of the Combined SIAS and LOOP Tests or any other test. RG 1.9 does specify that "Design provisions should include the capability to test each emergency diesel generator unit independently of the redundant units. Test equipment should not cause a loss of independence between redundant diesel generator units or between diesel generator load groups." However, this does not prohibit simultaneous testing.

2) NYPA's position is that simultaneous DG testing during a LOOP/LOCA test is performed in a manner that does not compromise safety and, therefore, within the provisions of ITS LCO 3.0.2 which allows intentionally relying on the ACTIONS for performance of Surveillances because of the following:

a) This test is conducted in Mode 5 or 6 when there are minimal requirements for AC sources, there is no requirement for redundant ESF systems, and manual initiation of ESF systems is permitted. However, the test can only be initiated when all DGs are available, all three safeguards power trains are connected to an Operable offsite source, and there is redundant decay heat removal capability.

b) This test is conducted with the plant deliberately configured to tolerate the potential for a loss of all AC power prior to initiation of the test by meeting the Required Actions of LCO 3.8.2 for no Operable offsite circuits and no Operable DGs. This is very conservative because with the plant shutdown there is sufficient time to terminate the test and manually align and operate any AC sources and/or ESF equipment required to respond to an event. Therefore, AC sources and ESF systems are fully functional even if not technically Operable.

c) When this test is in progress, the plant is restricted from performing any activity that is a precursor to an event that requires AC power for mitigation (i.e., fuel handling accident or inadvertent draining of the reactor coolant system).

d) NYPA also believes that an unplanned event (i.e., interaction between safeguard power trains) during the test is unlikely to result in damage to all three safeguards power trains such that at least one of the safeguards power trains could not be re-energized immediately from either one of the 3 DGs or one of the two circuits that connect safeguards power trains to the offsite circuits.

Finally, IP3 has extensive experience conducting this test on all three safeguards power trains simultaneously and has less potential for unidentified interactions than plants which never perform this test simultaneously especially when considering that the 10 year test in NUREG-1431 and RG 1.108 and RG 1.9 do not require that DG output breakers close and energize the associated busses and equipment (i.e., this test will not identify adverse interactions between safeguard power trains).

Therefore, NYPA believes that simultaneous testing of all three DGs during a LOOP/LOCA test has significant safety benefit and can be performed in performed in a manner that does not compromise safety.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--03**

RAI STATEMENT:

--CTS 4.6.A.4
--DOC LA.4

The CTS requirement to inspect the DGs is proposed to be relocated to the FSAR. Relocation of the requirement is acceptable, but the staff questions if the FSAR is the appropriate place for relocation.
Comment: The primary purpose of the FSAR is to be describe the plant and its conformance to regulations. Including specific requirements such as DG inspections in this document does not seem to be appropriate. The Technical Requirements Manual or similar document would be better suited to this purpose. The licensee is required to reconsider relocating this CTS requirement to the FSAR and to state how the relocated requirement is controlled.

NYPA RESPONSE:

Note that CTS requirements relocated to the FSAR to ensure that 10 CFR 50.59 applies to the requirement but the requirement will be implemented by plant procedures (which reference the FSAR).

NYPA has developed criteria for determining what goes into the FSAR and what goes into the TRM. Basically, the TRM is being reserved for those circumstances requiring action by control room operators. Examples of items designated for the TRM are requirements for more frequent monitoring if an indicator is not functional or a requirement to take action if river water level or temperature exceeds limits. Examples of items destined for the FSAR are requirements for the number and location of incore thimbles required when performing a flux map. Since the flux map will always be performed by a reactor engineer using a procedure, placing the requirement in the FSAR and the associated implementing procedure provides adequate assurance that the requirement will be met. The procedure reference to the FSAR provides assurance that the procedure is not changed unless the FSAR is revised in accordance with 10 CFR 50.59.

Based on this criteria, the requirement in CTS 4.6.A.4 that DGs be "inspected and maintained following the manufacturer's recommendations for this class of stand-by service" goes into the FSAR. Administrative Procedure AP-22.3. "Emergency Diesel Generator Inspection and Maintenance Schedule" is currently the implementing procedure for this activity.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--04**

RAI STATEMENT:

--ITS 3.8.1 Required Action A.3

This Required Action A3 is included in Condition A, "One Offsite circuit Inoperable." The staff questions whether or not this is entirely correct. Required Acton A.3 is invoked when the automatic transfer of 6.9 kV buses 1, 2, 3, and 4 to 6.9 kV buses 5 and 6 is disabled. The automatic transfer is disabled when offsite power is being supplied by the 13.8 kV source. However, when the automatic transfer is disabled, access to both offsite sources is not available to ESF buses 2A and 3A. In the staff's view, disabling the automatic transfer results in loss of 2 offsite circuits, not just one.

Comment:--Required Action A3 should be limited to those cases where only one offsite circuit is inoperable and an ESF bus is without offsite power, if any such conditions could exist. Also the wording "inoperable" should follow the word "Declare" as in the STS to avoid misinterpretation of the Action.

NYPA RESPONSE:

Note: RAI 3.8.1-04, 3.8.1-05, 3.8.1-06, 3.8.1-15 and 3.8.1-17 are all related to the following issues:

1-- The IP3 design has one immediate access and one delayed access offsite source and this design is consistent with requirements in GDC 17 as clarified in Reg. Guide 1.32, Rev.1.

2-- Allowable Out of Service Times (AOTs) established by RG 1.93 apply only to plants with 2 immediate access offsite sources.

3— To apply the RG 1.93 AOTs to a plant with a delayed access offsite source, additional compensatory measures are needed. These compensatory measures are found in LCO 3.8.1, Required Action A.2, and in the Note to Required Actions D.1 and D.2. (Note that STS Required Action A.2 is ITS Required Action A.3)

3.a-- Specifically, STS Required Action A.2 (one offsite circuit inoperable), requires declaring redundant required features inoperable if any safeguards power train has no offsite power. IP3 modified Required Action A.2 and its Completion Time to specify 'automatically available' to ensure it is understood that a delayed access circuit does not satisfy the requirement for offsite power for a safeguards power train being powered from the main generator via the UAT.

3.b-- Similarly, Required Actions D.1 and D.2 (one offsite circuit and one DG inoperable), requires entry in LCO 3.8.9 for any safeguards power train with no AC power source. IP3 modified the Note to Required Actions D.1 and D.2 for ITS Rev 0 by inclusion of the terms 'offsite or DG'. NYPA further modified the Note in Rev 1 to include the terms 'automatically available. This will ensure that it is understood that a delayed access circuit does not satisfy the requirement for offsite power for a safeguards power train being powered from the main generator via the UAT.

The following portion of the response is specific to RAI 3.8.1-04:

The Bases for NUREG-1431, LCO 3.8.1, states that "Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features." A plant with 2 immediate access circuits would never be in this position because if a train cannot be powered from an offsite source then two offsite circuits are inoperable and Required Actions C.1 and C.2 apply. It is apparent that STS Required Action A.2 is intended to address plants with a delayed access circuit and that STS Required Action A.2 applies to any train for which there is no immediate access offsite source available.

Required Action A.2 provides appropriate additional compensatory actions such that the AOTs from RG 1.93 can be applied to a plant with one immediate access offsite circuit and one delayed offsite circuit.

IP3 modified STS Required Action A.2 and its Completion Time to specify 'automatically available' to ensure it is understood that a delayed access circuit does not satisfy the requirement for offsite power for a safeguards power train being powered from the main generator via the UAT.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--05**

RAI STATEMENT:

--ITS 3.8.1 Condition C

See Q 3.8.1-04, regarding what constitutes the loss of 2 offsite sources.

Comment: The licensee to provide examples and describe circumstances that would constitute the loss of 2 offsite.

NYPA RESPONSE:

Note: RAI 3.8.1-04, 3.8.1-05, 3.8.1-06, 3.8.1-15 and 3.8.1-17 are all related; refer to RAI 3.8.1-04 reply for general discussion.

The following portion of the response is specific to RAI 3.8.1-05:

Two offsite circuits are inoperable when both the immediate access circuit and the delayed offsite circuit are not available to one or more safeguard power trains.

The LCO section of the IP3 3.8.1 Bases includes a detailed description of the offsite circuits including that portion of the circuit that is common to both the immediate access and delayed access circuits. The most probable cause of two inoperable offsite circuits is a failure in a portion of the circuit that is common to both offsite circuits.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--06**

RAI STATEMENT:

--ITS 3.8.1 Condition D

The staff has a question regarding under what circumstances there could be a loss of one offsite source and one DG inoperable that results in an ESF bus being without AC power.

Comment: Provide examples which would describe the above circumstances

NYPA RESPONSE:

Note: RAI 3.8.1-04, 3.8.1-05, 3.8.1-06, 3.8.1-15 and 3.8.1-17 are all related; refer to RAI 3.8.1-04 reply for general discussion.

The following portion of the response is specific to RAI 3.8.1-06:

NYPA position is that the Note to LCO 3.8.1, Required Actions D.1 and D.2. should be interpreted as "no 'immediate access' AC power source to any train" because this interpretation ensures that the Note provides appropriate compensatory action that allows the AOTs in RG 1.93 to apply to a plant with one immediate access and one delay access offsite source.

NYPA modified the Note to LCO 3.8.1, Required Actions D.1 and D.2. in NUREG-1431 to include the phrases in single quotation marks so that the Note reads as follows: "Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition D is entered with no 'offsite or DG' AC power source to any train." The IP3 Note differs from the NUREG-1431 Note by the inclusion of the term 'offsite or DG'. NYPA further modified the Note for ITS Rev 1 to read; "no offsite or DG AC power source *'automatically available'* to any train." This will ensure that it is understood that a delayed access circuit does not satisfy the requirement for offsite power for a safeguards power train being powered from the main generator via the UAT.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--07**

RAI STATEMENT:

--NUREG Condition F

The licensee has not included sequencers in ITS LCO 3.8.1 and, consequently, NUREG Condition F is proposed for deletion.
Comment: The justification for this appears to be that IP3 uses individual load timers instead of sequencers. However, DOC L.6 includes a discussion of the load timers and how they affect both the DGs and the offsite power if the timer is inoperable. This is reflected in the proposed Note to ITS SR 3.8.1.11 which states that the load timers are not required to be OPERABLE if the associated equipment has the automatic initiation capability disabled. Stated differently, an inoperable load timer can be compensated for by disabling the associated equipment. Since an inoperable load timer has an impact on both offsite and onsite power, some action is required when the timer is found to be inoperable. This action should be a requirement in TS; i.e., a Condition of one or more load timers inoperable with a Required Action to disable associated equipment within a specified period of time. Load timers may also be required to be included in the LCO.

NYPA RESPONSE:

NYPA believes that the presentation of requirements provided in the IP3 ITS conversion submittal is acceptable and more appropriate for IP3 which has individual load time delay relays (versus a safeguards bus sequencer). Implicit in the note in NUREG-1431 requiring retention of Condition F is the assumption that all loads in a safeguard power train remain inoperable until the sequencer is restored to Operable. This is not true at plants with individual load time delay relays. The IP3 approach eliminates redundancy and potential contradictions concerning which LCO governs individual loads affected by an inoperable time delay relay because load time delay relays can fail in two ways: an individual load starts outside its design interval (i.e., potential impact on DG and offsite source) or an individual load fails to start (i.e., no impact on DG and offsite source). This is particularly relevant because IP3 load time delay relays are verified to satisfy design interval requirements by disabling automatic initiation capability of the load and removing and bench testing the relay. This is necessary because time delay relays must be verified at 18 month intervals (versus 24 months) for the LOOP/LOCA test. Therefore, IP3 will use this Note to avoid unnecessary entry into the Actions for an inoperable DG and offsite source every time the SR is performed. An expanded discussion regarding load timers has been added to the Bases for SR 3.8.1.11.

The IP3 presentation maintains requirements identical to the requirements that would be imposed if NUREG-1431, Condition F, was used because ITS SR 3.0.1 ensures that Condition

D (one offsite source and one DG inoperable) if the plant is ever operated with a time delay relay not within the required design interval.

Final Note: NYPA can develop a case that the note in NUREG-1431 allowing deletion of Condition F is applicable to IP3 because NYPA analysis indicates that the starting load overlap created by the failure (early or late start) of any individual load time delay relay will not result in either the DG or the offsite source exceeding any design limits. A DG or offsite circuit will not exceed design limits unless more than one time delay relay fails and results in an overlap start of three loads.

The Bases for SR 3.0.1 states that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification ensures that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--08**

RAI STATEMENT:

--ITS SR 3.8.1.7

The Note prohibiting this SR from being performed in Modes 1 or 2 is proposed to be deleted. However, no justification has been provided.

Comment: The licensee is requested to provide a detailed discussion of how this power transfer is accomplished and why it is safe to do this in Modes 1 and 2. The discussion should include a system description and cover such things as system impedance, voltages, circulating currents, and bus' ampacities associated with the transfer.

NYPA RESPONSE:

NYPA revised ITS SR 3.8.1.7 to include the STS Note that prohibits manual transfer of AC power sources from the normal offsite circuit to the alternate offsite circuit in Modes 1 and 2.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--09**

RAI STATEMENT:

--Insert 3.8-8-01 (ITS SR 3.8.1.8)

The staff does not understand Notes 1 and 2 associated with this SR. The requirement is to demonstrate the automatic transfer every 24 months. However, this need not be done if the 138 kV source is not powering Bus 5 and 6, or the Unit Aux Transformer is not powering Bus 2 and 3, according to Note 2. In this case, it is entirely possible for Note 2 to completely supercede the SR. This is not acceptable. In a practical sense, this SR will never be performed because the Unit Aux Transformer will not be energized in Modes 3-6.

This brings up a question regarding Note 1. Since the Unit Aux Transformer will not be energized in Modes 3-6, and since the purpose of this SR is to demonstrate automatic transfer from the Unit Aux transformer to offsite power, it appears that Note 1 needs to be changed. It should read something like "This SR should not be performed in Modes 1 and 2 above [10]% power."
Comment: The licensee is required to address staff concerns regarding Notes 1 and 2.

NYPA RESPONSE:

This RAI is related to RAI 3.8.1-18. Part 2 of this RAI, related to changes to IP3 SR 3.8.1.8, Note 1, is fully addressed in RAI 3.8.1-18.

IP3 ITS SR 3.8.1.8 requires verification of the automatic transfer of AC power for 6.9 kV buses 2 and 3 from the unit auxiliary transformer to 6.9 kV buses 5 and 6. This feature is only required to be operable when the main generator is supplying safeguards power train 2A/3A and 138 KV is the immediately available offsite circuit. (Note that safeguards power trains 5A and 6 A are always powered from an offsite source. However, IP3 has the option of transferring the power source for reactor coolant pumps 1, 2, 3, and 4 to the main generator and when this lineup is established, safeguards power train 2A/3A is also powered from the main generator. This creates the need for the auto transfer of safeguards train 2A/3A to the offsite source.)

Note 2 to ITS SR 3.8.1.8 states that verification of the auto transfer function is "Only required to be met if 138 kV offsite circuit is supplying 6.9 kV bus 5 and 6 and the Unit Auxiliary Transformer is supplying 6.9 kV bus 2 or 3" (i.e., the main generator is supplying safeguards power train 2A/3A).

Note 2 to IP3 ITS SR 3.8.1.8 is needed because the Applicability of the auto transfer function is different from the applicability of the offsite circuit. Without Note 2, the SR for auto transfer function would be required by SR 3.0.1 to be performed and met even when the plant configuration prevents the function from being Operable and function is not needed.

Assurance that the autotransfer SR will be performed and met prior to entry into the Applicable Mode or Condition (i.e., main generator status) is provided by SR 3.0.4 which requires that an SR is performed within Frequency and met prior to entering the Applicable Mode or Condition. Staff concerns related to Note 1 (i.e., the conditions under which IP3 ITS SR 3.8.1.8 must be performed are addressed in RAI 3.8.1-18.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--10**

RAI STATEMENT:

--ITS SR 3.8.1.9

The licensee has proposed to delete the Mode restriction Note but has not provided a justification.
Comment: The licensee is requested to provide a discussion regarding how this SR can be safety performed at power, or retain the Mode restriction Note.

NYPA RESPONSE:

NYPA revised ITS SR 3.8.1.9 (STS SR 3.8.1.13) to include STS Note that prohibits performing an SR in Modes 1 and 2 that verifies that DG trips are bypassed on an ESFAS signal.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--11**

RAI STATEMENT:

--ITS SR 3.8.1.10

The licensee has proposed to delete the Mode restriction Note but has not provided a justification
Comment: The licensee is requested to provide a justification for deleting the Mode restriction Note in the SR, or retain the NUREG restriction.

NYPA RESPONSE:

NYPA revised ITS SR 3.8.1.10 (STS SR 3.8.1.14) to include the STS Note that prohibits performance of the DG endurance run in Modes 1 and 2.

This item is a duplicate of RAI 3.8.1-1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--12**

RAI STATEMENT:

--ITS SR 3.8.1.11
--STS SR 3.8.1.18 Note

a) The Note for STS SR 3.8.1.18 states, "This Surveillance shall not be performed in Mode 1, 2, 3, or 4." This Note has not been adopted in corresponding ITS SR 3.8.1.10.
Comment: No justification has been provided to support this proposed difference. Revise the submittal to provide the appropriate justification for the proposed difference, or conform to the STS.

b) What is the required design interval for the load timers? The interval is not included in the SR or in the Bases.
Comment: Something needs to be added to the submittal that establishes what the intervals are or where they can be found.

NYPA RESPONSE:

a) IP3 SR 3.8.1.11, verification that time delay relays function within the required design interval, has a Frequency of 18 months (versus 24 months) for the LOOP/LOCA test. IP3 load time delay relays are verified to satisfy design interval requirements by disabling automatic initiation capability of the load and removing and bench testing the relay. While the timer is removed, automatic initiation capability of the component is blocked and there is no potential that a load start outside the required design interval could cause either the DG or offsite circuit to exceed design limits.

b) Load timer design intervals are currently maintained in the load timer calculation and the SR implementing procedure.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--13**

RAI STATEMENT:

--ITS SR 3.8.1.12 Note 3

The proposed Note 3 to ITS SR 3.8.1.12 allows the SR to be conducted on all three DGs at the same time. This SR must be conducted in Modes 5 or 6 when 2 of the 3 DGs are required to be OPERABLE.
Comment: See staff's comment regarding proposed Note 3 in Q 3.8.1-02.

NYPA RESPONSE:

See Response to RAI 3.8.1-02.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--14**

RAI STATEMENT:

--ITS SR 3.8.1.13 - Insert 3.8-16-01

Conducting this SR on the DGs required to be OPERABLE could cause electrical system perturbations and challenges to safety systems. The licensee is encouraged to revise the Note to limit the SR to one DG at a time, consistent with NUREG-1431.
Comment: The proposed Note 2 may have to be changed depending on the results (i.e.; responses from licensee) from review of Q 3.8.1-02 and Q 3.8.1-12.

NYPA RESPONSE:

See Responses to RAI 3.8.1-02.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--15**

RAI STATEMENT:

--Bases Page B 3.8-5 - Action A.1 and A.2 (Insert B3.8-5-01)

The IP3 offsite power system is somewhat complex, and the Bases for this Action could be improved by adding a discussion of what system conditions results in a loss of one offsite source. For example, loss of the Station Aux Transformer would appear to constitute a loss of one offsite source, but the failure of one or more of the 4 SSTs would constitute a loss of both offsite circuits to one or more ESF busses.
Comment: The licensee should consider expanding this Bases section. Of particular interest would be a discussion regarding why the licensee does not consider the offsite circuits to be inoperable when the automatic transfer capability is disabled.

NYPA RESPONSE:

Note: RAI 3.8.1-04, 3.8.1-05, 3.8.1-06, 3.8.1-15 and 3.8.1-17 are all related.

This RAI is fully addressed in the responses to RAI 3.8.1-04, 3.8.1-05 and 3.8.1-06,

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--16**

RAI STATEMENT:

--Bases Page B3.8-6 - Action A.3

This Bases discussion needs to be revised to more accurately reflect the IP3 design. Specifically, the bases needs to be revised to reflect the fact that there are 4 ESF buses, and that when the 13.8 kV offsite source is being used, the automatic transfer to offsite may well be blocked for 2 of the 4 buses. The remaining two buses must be evaluated for inoperable redundant features, not just one bus.
Comment: These are changes that need to be made to make the bases correct.

NYPA RESPONSE:

IP3 nomenclature and practice are that there are 3 safeguards power trains. The three safeguards power trains are train 5A (480 volt bus 5A and associated DG 33), train 6A (480 volt bus 6A and associated DG 32), and train 2A/3A (480 volt buses 2A and 3A and associated DG 31). This configuration is explained in the Background section of the Bases.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--17**

RAI STATEMENT:

--Bases Page B3.8-13 - Actions D.1 and D.2

What is the intent of this Bases discussion?
Comment: When the 13.8 kV source is being used to supply offsite power Bus 5A and Bus 6A and the automatic transfer feature to supply offsite power to Bus 2A and Bus 3A is disabled, what condition is the plant in? Is LCO 3.8.9 required to be entered?

NYPA RESPONSE:

Note: RA 3.8.1-04, 3.8.1-05, 3.8.1-06, 3.8.1-15 and 3.8.1-17 are all related; refer to RAI 3.8.1-04 reply for general discussion.

The following portion of the response is specific to RAI 3.8.1-17:

When the 13.8 kV source is being used to supply offsite power Bus 5A and Bus 6A and the automatic transfer feature to supply offsite power to Bus 2A and Bus 3A is disabled, there is an immediate access circuit available to safeguards power train 5A and 6A but only a delayed access source available to safeguards power train 2A/3A. Therefore, there is only one offsite circuit inoperable (i.e., the immediate access circuit to safeguards power train 2A/3A. IP3 is in Condition A.

In this situation, offsite power is available to safeguards power train 2A/3A as soon as the operator verifies that the reactor coolant pumps have tripped (to ensure the 13.8 kV circuit is not overloaded) and manually closes the breakers. As discussed in the response to RAI 3.8.1-04, IP3 LCO 3.8.1, Required Action A.3, provides sufficient compensatory measures so that the RG 1.93 AOTs for one inoperable offsite source are appropriate in this situation.

In the same situation but with the DG that supports 2A/3A inoperable, Condition D is entered. The Note to Required Actions D.1 and D.2 requires that you enter LCO 3.8.9 even though the 2A/3A buses are energized because if the main generator trips then the 2A/3A bus is de-energized.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--18**

RAI STATEMENT:

--Bases Page B3.8-20 - Insert B3.8-20-01

The licensee is required to provide additional discussion on how ITS SR 3.8.1.8 is conducted.

Comment: The staff is particularly interested in how a transfer scheme that functions on low voltage can be adequately tested without incurring the actual low voltage.

NYPA RESPONSE:

This RAI is related to RAI 3.8.1-09.

SR 3.8.1.8 is a verification that 6.9 kV buses 2 and 3 will auto transfer (fast transfer) from the Unit Auxiliary transformer to 6.9 kV buses 5 and 6 (i.e. station auxiliary transformer) following a loss of voltage on 6.9 kV buses 2 and 3 following a trip of the main generator.

Options for performing a test of this feature are 1) tripping the main generator at a low power level during a reactor shutdown (which the NRC staff appears to be recommending in RAI-3.8.1-09), or 2) a "bench" test.

Currently, IP3 CTS do not require testing this feature although the feature is tested when the reactor is shutdown without deliberate initiation of the transfer. This is identical to the approach specified in the Bases for HB Robinson ITS SR 3.8.1.15 which appears to test the identical feature. It is also very important to note that this feature is fully tested every time the reactor or main generator trips from above approximately 5% RTP.

As stated in the Bases for ITS SR 3.8.1.8. an actual demonstration of this feature requires the tripping the main generator while the reactor is at power with the main generator supplying 6.9 kV buses 2 and 3. This will cause perturbations to the electrical distribution systems that could challenge unit safety systems during a plant shutdown. Therefore, in lieu of actually initiating a circuit transfer, testing that adequately shows the capability of the transfer is acceptable. This transfer testing may include any sequence of sequential, overlapping, or total steps so that the entire transfer sequence is verified.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.1 AC Sources - Operating**

NRC RAI No: **3.8.1--19**

RAI STATEMENT:

--Bases Page B3.8-32 - Insert B3.8-32-01, B3.8-32-02

The licensee is requested to provide a discussion regarding the intent of the second paragraph of the insert.

Comment: Does this paragraph mean that the licensee will deliberately enter the Actions of LCO 3.8.2 by making all 3 DGs inoperable in order to conduct this SR on all DGs at the same time? If not, what does it mean? Is this provision part of the CTS? The proposed Note 2 for SR 3.8.1.13 is a subset of the above staff's question.

NYPA RESPONSE:

NYPA's position is that simultaneous DG testing during a LOOP/LOCA test is performed in a manner that does not compromise safety and, therefore, within the provisions of ITS LCO 3.0.2 which allows intentionally relying on the ACTIONS for performance of Surveillances. See Responses to RAI 3.8.1-02.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.2 AC Sources - Shutdown**

NRC RAI No: **BSR 3.8.2--01**

RAI STATEMENT:

BSR--CTS 3.7.F
--DOC L.1

For Information Only:
CTS 3.7.F requires AC power under all conditions. In DOC L.1, the licensee attempts to make the case that ITS LCO 3.8.2 is not applicable when the reactor is defueled.
Comment: The proposed change appears to be a beyond scope issue that will have to be addressed independent of the conversion review.

NYPA RESPONSE:

ITS 1.0 includes the definition: A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant loop temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1, "with fuel in the reactor vessel." (emphasis added).

The Applicability for ITS 3.8.2 is "Modes 5 and 6 and during movement of irradiated fuel assemblies" which is different from the equivalent CTS Applicability which is "at all times." Based on the ITS 1.0 definition of Mode, ITS LCO 3.8.2 does not apply when there is no fuel in the reactor vessel and there is no movement of irradiated fuel in progress. NYPA interprets this to mean that if all of the fuel on site is in the spent fuel pit or the new fuel storage pit, then there is no Technical Specification governing operability of AC sources. This is consistent with 10 CFR 50.36 criteria governing Technical Specifications. DOC L.1 justifies the change in applicability. NYPA does not believe that this is a beyond scope change

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.2 AC Sources - Shutdown**

NRC RAI No: **3.8.2--01**

RAI STATEMENT:

--CTS 3.7.F
--DOC M.3

The staff is not certain what is meant by the discussion of LCO 3.0.2 in DOC M.3. Does this mean that the requirements of LCO 3.8.2 can be not met for testing purposes provided the Required Actions of LCO 3.8.2 are implemented prior to the testing?

Comment: If this is the case, the licensee is requested to provide a discussion on how DOC M.3 is consistent with the NUREG Bases discussion regarding entry into the Actions in a manner that does not compromise safety. Also, how is the discussion for DOC M.3 consistent with DOC L.2 as it relates to performance of SRs in Modes 5 & 6.

NYPA RESPONSE:

DOC L.3 justifies the Note in SR 3.8.2.1 that selected LCO 3.8.1 SRs must be met but do not have to be performed to demonstrate DG Operability when in Modes 5 and 6. DOC L.3 paraphrases the NUREG-1431 Bases by stating that the reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude reenergizing a required 480 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG.

An implicit assumption in this justification in Bases for SR 3.8.2.1 is that only the minimum number of offsite sources (i.e., one offsite source to one or two safeguards power trains) and minimum number of DGs (i.e., two) are Operable and that the SRs are performed on these components at a time when redundant components are unavailable. Therefore, a test induced failure could cause of loss of all AC without the option of using redundant components. Additionally, NUREG-1431 does not require deliberately configuring the plant to tolerate the potential for a loss of all AC power prior to initiation of the test and does not restrict the plant from performing any activity that is a precursor to a shutdown event that requires AC power for mitigation.

Conversely, concurrent testing of DGs during the LOOP/LOCA test does not compromise safety because: a) the test can only be initiated when all DGs are Operable and there is full redundancy for all ESF systems; b) the plant is deliberately configured to tolerate the potential for a loss of all AC power prior to initiation of the test; and, c) the plant is restricted from performing any activity that is a precursor to a shutdown event that requires AC power for mitigation. NYPA also believes that an unplanned event during the test is unlikely to result in damage to all three safeguards power trains such that at least one of the safeguards power trains could not be re-energized immediately from either one of the 3 DGs or one of the two circuits that connect safeguards power trains to the offsite circuits.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.2 AC Sources - Shutdown**

NRC RAI No: **3.8.2--02**

RAI STATEMENT:

--Bases Page 3.8-37 - Insert B3.8-37-01

This Bases insert adds the provision that safeguards power trains may be cross connected in Modes 5 and 6.
Comment: This is acceptable. However, the Bases should be expanded to include a discussion of how this cross connection is accomplished, and any restrictions there may be regarding cross connection, such as not connecting 2 DGs to the same bus.

NYPA RESPONSE:

NYPA revised ITS so that the LCO section of the Bases specifies that interlocks which disconnect 480 V buses before DGs are automatically connected to the bus must be Operable.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.2 AC Sources - Shutdown**

NRC RAI No: **3.8.2--03**

RAI STATEMENT:

--Bases Page B3.8-38 - LCO

The last paragraph of the NUREG LCO Bases includes the provision that safeguards trains may be cross tied to allow a single offsite to power all required trains. The NUREG Bases are intended to address a cross tie upstream of the actual safeguards buses. In the ITS Bases, the NUREG Bases are modified to state that the "Safeguards power" trains may be cross tied. For IP3, this has a potentially different meaning than was intended in the NUREG. At IP3, it appears that the safeguards buses can be cross tied at the bus level which would allow offsite power to be fed through one bus to another. This was not the intent of the NUREG. Comment: Is the above the intent of the ITS Bases discussion? If not, the Bases should be revised to clearly identify the intent.

NYPA RESPONSE:

See Response to RAI 3.8.2-02.

NUREG-1431, LCO 3.8.2, Bases state: "It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains." NYPA believes that this is intended to allow safeguards buses to be cross tied at the bus level when shutdown. This is an explicit relaxation of the stipulation in NUREG-1431, LCO 3.8.1, Bases statement: "The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.2 AC Sources - Shutdown**

NRC RAI No: **3.8.2--04**

RAI STATEMENT:

--Bases Page B3.8-40 - Insert B3.8-40-01.

What the insert means, what its intended purpose is, and why this insert is considered more appropriate than the SR 3.8.2.1 Bases material proposed for deletion.
Comment: The licensee is requested to provide a discussion of the above staff's concern.

NYPA RESPONSE:

NYPA revised the ITS to remove Insert B3.8-40-01 and restore the discussion in NUREG-1431.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.2 AC Sources - Shutdown**

NRC RAI No: **3.8.2--05**

RAI STATEMENT:

--JFD DB.1
--ITS 3.8.2 Condition B
--STS 3.8.2 Condition B
--Bases for Required Action B.1 for ITS 3.8.2, STS Bases markup
--page B 3.8-39, insert B 3.8-39-01
--Bases for Required Actions B.2.1, B.2.2, B.2.3, and B.2.4 for ITS 3.8.2,
--STS Bases markup page B 3.8-39, insert B 3.8-39-01

In the event that both required DGs are inoperable, Condition B for ITS 3.8.2 would allow continuation of Core Alterations, movement of irradiated fuel assemblies, and positive reactivity additions. Condition B for corresponding STS 3.8.2 does not allow those actions in the event that the required DG is inoperable. The Bases for Required Actions B.2.1, B.2.2, B.2.3, and B.2.4 for ITS 3.8.2 states, "Therefore, with two required DGs inoperable, it is required to suspend Core Alterations, movement of irradiated fuel assemblies, and operations involving positive reactivity additions."

Comment: There is an apparent discrepancy between the Required Actions associated with Condition B and the Bases. Revise the submittal to resolve the discrepancy. Additionally, JFD DB.1 does not explain why the proposed difference between the STS and ITS is acceptable. Revise the submittal to explain why the proposed difference is acceptable, or delete proposed Required Action B.1 for ITS 3.8.2.

NYPA RESPONSE:

NYPA has revised ITS to delete the Required Action B.1 proposed in Revision 0. The need for this provision has been eliminated by the incorporation of License Amendment 194 that was isssued after ITS Revision 0 was submitted. Amendment 194 identifies specific conditions for which only 1 DG is sufficient to support plant operations in Modes 5 and 6 and during movement of irradiated fuel assemblies. LCO 3.8.2.c has been added, Required Action B.1 was deleted, and corresponding Bases were revised to address this RAI and to incorporate Amendment 194.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--01**

RAI STATEMENT:

--CTS 3.7.A
--DOC L.3

A change to CTS to add SR 3.8.3.5 and Condition G is included in the CTS markup. The justifications for this change are DOC M.3 and DOC L.3. However, no DOC L.3 is included in the submittal.
Comment: The licensee is requested to provide this DOC.

NYPA RESPONSE:

NYPA has revised ITS 3.8.3 to add DOC L.3. The less restrictive change to add a 48-hour completion time is acceptable because air receiver pressure sufficient to support four start attempts contains substantial margin before reaching a condition that would prevent the DG from performing its safety function. Therefore, if sufficient starting air for at least one start attempt is maintained during the new restoration period then the DG is still capable of performing its safety function. This change has no significant impact on safety because of the limited level of degradation permitted by this new condition and the limited time this condition is allowed to persist.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--02**

RAI STATEMENT:

--ITS LCO 3.8.3 - Condition B

It is not clear to the staff how this Condition is supposed to work. The requirement is for 5891 gal. in all storage tanks, but it is not clear how it is assured that the fuel will be available in the tank(s) associated with the DG(s) required to be OPERABLE.
Comment: The Bases discussion of this Condition does not provide clarification of this issue. This staff concern is also applicable to proposed ITS SR 3.8.3.2b; i.e., the SR does not ensure that fuel oil is available in the tank(s) associated with the DG(s) required OPERABLE.

NYPA RESPONSE:

A low level in the day tank for any DG will open the fill valves for that day tank and start the pump in the associated day tank. Once started by low level in the associated day tank, the transfer pump will continue to run until that day tank is filled. However, any operating transfer pump will fill any day tank with a normal or emergency fill valve that is open. When a day tank is at approximately 158 gallons (90% full), a switch initiates closing of the day tank normal and emergency fill valves and stop the associated transfer pump.

Oil in a storage tank associated with a particular DG will fill the day tanks associated with the other two DGs. However, this will only happen if a low level in the day tank associated with the storage tank starts the pump and a low level in the day tank of the non associated DGs opens the fill valves for its day tank.

There is no assurance that the day tank for a DG required to be Operable will refill automatically from a storage tank not associated with the Operable DG. Therefore, NYPA revised ITS 3.8.3, Condition B, and SR 3.8.3.2.b to specify that the 'required volume' of DG fuel oil in underground tanks must be divided between the tanks associated with the DGs required to be Operable.

Note that the 'required volume' as stated in the proposed ITS Revision 0 was 5891 gallons. Recent calculations by NYPA have changed the required volume to 5365 gallons. Similarly, the 'required volume' for the reserve storage tanks changed from 30026 gallons to 26826 gallons. Therefore, NYPA is incorporating these revised volumes in proposed ITS Revision 1 as supported by new DOCs L.4 and L.5.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--03**

RAI STATEMENT:

--JFD PA.1
--ITS SR 3.8.3.6
--STS SR 3.8.3.5

STS SR 3.8.3.5 requires to, "Check for and remove accumulated water from each fuel storage tank." Corresponding ITS SR 3.8.3.6 requires to, "Check for and remove accumulated water from each underground fuel storage tank."
Comment: ITS SR 3.8.3.6 does not address the reserve storage tank(s), and JFD PA.1 does not explain why it is acceptable to exclude the reserve storage tank(s) from the surveillance. Revise the submittal to explain why it is acceptable to exclude the reserve storage tank(s) from the surveillance, or expand the SR to include the reserve storage tank(s).

NYPA RESPONSE:

IP3's DG offsite reserve fuel oil is stored by Consolidated Edison in tanks that are part of a network of tanks used to store oil used to support operation of several gas turbine peaking units. The tanks used to store IP3's reserve fuel oil are above ground or inside buildings and above grade. There are no existing programmatic requirements for periodic checking for and removal of water from these tanks because the tank internals are not easily accessible in a manner that permits performing this task.

NYPA believes this is acceptable because the presence of water in the offsite reserve tanks is not an Operability concern for the following reasons:

1) Quantities of water significant enough to interfere with DG operation due to entrainment are expected to be identified during the routine use of this oil for the gas turbine peaking units. Fuel oil in the reserve tanks is not automatically supplied to the DGs and must be transferred by truck from the reserve tanks to the onsite storage tanks. Any substantial amounts of water in the oil can be removed from the oil during this transfer and not added to the onsite storage tanks. Therefore, the presence of water in the offsite reserve tanks is not a threat to DG Operability due to entrainment.

2) Water in the offsite reserve tanks as a catalyst for microbiological oil degradation is not a significant concern because the fuel oil turnover rate is significantly higher than in a typical nuclear plant due to its use as gas turbine fuel. Additionally, the tanks are periodically monitored for particulate. Therefore, the presence of water in the offsite reserve tanks is not a threat to DG Operability in its role as a catalyst for microbiological oil degradation.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--04**

RAI STATEMENT:

--Bases Insert B3.8-41-01 (page 2)

The licensee is requested to provide more details on how the fuel oil transfer system is designed.

Comment: Specifically, how is fuel oil from a storage tank not associated with a particular DG made available to that DG when its day tank level is low? This discussion should also explain how SR 3.8.3.2.b works to ensure adequate fuel to all required DGs in Modes 5 & 6.

NYPA RESPONSE:

See Response to RAI 3.8.3-02.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--05**

RAI STATEMENT:

--Insert B 3.8-43-01 (Page 2), Insert B3.8-043-01 (Page 3)

On Page 2 of this insert it is stated that "sufficient fuel oil to support continuous operation while a fuel transfer from the offsite DG Fuel oil reserve ..." On Page 3 of this insert it is stated that "Condition C is only applicable in Modes 1, 2, 3 and 4 because the offsite DG fuel oil reserve is required to be available only in those Modes."
Comment: These two inserts appear to be in conflict with each other. The licensee should justify when the offsite fuel oil reserve is required and then revise these Bases, as necessary, to be consistent.

NYPA RESPONSE:

Insert B 3.8-43-01 (Page 2) is the Bases for IP3 ITS LCO 3.8.3, Condition B, which is the Condition governing insufficient cumulative oil volume in the onsite storage tanks in Modes 5 and 6 and when moving irradiated fuel. When in Modes 5 and 6 and when moving irradiated fuel only the onsite tank volume is required by tech specs. There is no requirement for a minimum volume in the offsite reserve tanks when in Modes 5 and 6.

The bases explain that the onsite storage tanks normally have sufficient oil to allow time to get additional oil "from the offsite DG fuel oil reserve or from another offsite source." This is not intended to imply that the offsite DG fuel oil reserve tanks are required but that they are a potential source of oil, in addition to fuel oil vendors in the area, once the volume of oil required by Technical specifications is depleted. Even when not required to support IP2 or IP3 operations, the offsite fuel oil reserve tanks are likely to contain a substantial amount of oil because these tanks are part of a network of tanks used to store oil to support operation of several gas turbine peaking units.

Insert B 3.8-43-01 (Page 3) is the Bases for IP3 ITS LCO 3.8.3, Condition C, which is the Condition governing insufficient oil volume in the offsite reserve storage tanks Modes 1 through 4.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--06**

RAI STATEMENT:

--Bases Pg. B3.8-44 - Insert B3.8-44-03

This Bases discussion addresses the offsite DG fuel oil reserve properties not being within the limits of ITS 5.5.12. However, Specification 5.5.12 does not include any limits for this fuel oil. Specification 5.5.12 requires this fuel to be a commercial grade and suitable for use in the DGs.

Comment: Is this Bases discussion addressing the properties of fuel oil required to be maintained by IP2 operators? If not, what limits are supposed to be addressed by this Bases discussion and associated LCO Condition?

NYPA RESPONSE:

NYPA revised the ITS conversion submittal so that Diesel Fuel Oil Testing Program requirements apply to both onsite DG fuel oil storage tanks and offsite DG reserve fuel oil storage tanks. Additionally, ITS 5.5.12 has been revised to conform to program requirements as specified in NUREG-1431 including TSTF-106 and 118. See response to RAI 5.5-05.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--07**

RAI STATEMENT:

--Bases for ITS SR 3.8.3.3, STS Bases markup page B 3.8-46,
--insert B3.8-46-03
--Bases for STS SR 3.8.3.3

The Bases for STS SR 3.8.3.3 specifies the tests, limits, and applicable ASTM standards for new fuel oil testing. This material has not been retained in the Bases for corresponding ITS SR 3.8.3.3. Instead the Bases for ITS SR 3.8.3.3 refers to the administrative program developed to meet ITS 5.5.12.
Comment: No justification has been provided to support this proposed difference. Revise the submittal to provide the appropriate justification for the proposed difference, or conform to the STS.

NYPA RESPONSE:

NYPA revised ITS so that Diesel Fuel Oil Testing Program requirements apply to both the onsite DG fuel oil storage tanks and the offsite DG reserve fuel oil storage tanks. Additionally, ITS 5.5.12 was revised to conform to program requirements as specified in NUREG-1431 including TSTF-106 and 118. See response to RAI 5.5-05.

NYPA will not identify specific ASTM standards, revisions and exceptions to those standards in the Bases for ITS SR 3.8.3.3 because this information is considered redundant to information that will be contained in the NYPA and Consolidated Edison programs being developed to conform to ITS 5.5.12. IP3 does not have any current FSAR, SER or Technical Specification requirements for diesel fuel oil testing. Therefore, not including this information in the ITS 3.8.3 is consistent with current licensing basis.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--08**

RAI STATEMENT:

--Bases for ITS SR 3.8.3.4, STS Bases markup page B 3.8-47,
--insert B3.8-47-06

The Bases for ITS SR 3.8.3.4 describes testing the fuel oil in reserve storage.

Comment: Confirm that this description applies to the fuel oil stored at the Buchanan Substation as well as at the Indian Point site.

NYPA RESPONSE:

NYPA revised ITS so that a Diesel Fuel Oil Testing Program with the same requirements must be developed and will apply to both the onsite DG fuel oil storage tanks and the offsite DG reserve fuel oil storage tanks controlled by Consolidated Edison. Additionally, ITS 5.5.12 was revised to conform to program requirements as specified in NUREG-1431 including TSTF-106 and 118. See response to RAI 5.5-05.

LCO 3.8.3 was revised to establish the following Conditions and Required Actions for both onsite and offsite DG fuel oil:

Condition D specifies requirements with "One or more DG fuel oil storage tanks or reserve storage tanks with fuel oil total particulates not within limits." Required Action D.1 specifies: "Restore stored fuel oil total particulates within limits." Completion Times are: "7 days for DG fuel oil storage tanks AND 30 days for reserve storage tanks"

Condition E specifies requirements with "One or more DG fuel oil storage tanks or reserve storage tanks with fuel oil properties other than particulates not within limits. Required Action E.1 is: "Restore stored fuel oil properties within limits." Completion Times are: "30 days for DG fuel oil storage tanks AND 60 days for reserve storage tanks"

The Revision 0 version of Condition F is deleted.

As a result of this change, IP3 will differ from the NUREG only in that additional time is provided to correct out-of-specification conditions in the offsite reserve tanks and Condition E specifies "oil properties other than particulates" rather than new fuel oil properties.

The first difference is acceptable because this fuel oil is not required to be supplied to the DGs until 48 hours after event initiation. Additionally, one or more of a multiple number of tanks may be used to satisfy reserve fuel oil storage requirements.

The second difference is needed in recognition of Consolidated Edison's practice of sampling tanks for ITS 5.5.12.b requirements after the fuel oil addition versus using a sample the new fuel. This is discussed in the responses to RAIs for Section ITS 5.5.12.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--09**

RAI STATEMENT:

--Bases for ITS SR 3.8.3.6, STS Bases markup page B 3.8-48,
--insert B3.8-48-02
--Bases for STS SR 3.8.3.5

The Bases for STS SR 3.8.3.5 states, "The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of the Surveillance." The Bases for corresponding ITS SR 3.8.3.6 states, "Unless the volume of water is sufficient that it could impact DG Operability, the presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed within 30 days of the performance of the Surveillance."
Comment: No justification has been provided to support the proposed difference. Revise the submittal to provide the appropriate justification for the proposed difference, or conform to the STS.

NYPA RESPONSE:

Water entrainment in the fuel oil is an Operability concern only if there is a sufficient amount of water such that water level approaches tank suction level. This is an immediate Operability concern and the DG should be considered inoperable immediately if the water is not removed as part of the surveillance that detects the water in the tank.

Water as a catalyst for microbiological oil degradation is not an immediate DG Operability concern because microbiological oil degradation is a gradual process and is tracked as part of the SRs governing fuel oil particulate. Therefore, a small amount of water that is not a direct threat to DG Operability should not have to be removed immediately in order to assure DG Operability. At IP3, opening a tank and staging for water removal is a significant evolution that cannot be completed as part of the process used to detect water.

NYPA revised ITS so that the IP3 ITS Bases for SR 3.8.3.6 specify that 7 days (versus 30 days in the original submittal) is formally recognized as a reasonable time to remove water from the tank if the amount of water in the tanks is not considered a direct threat to DG Operability. This will allow water removal to be performed as a non-emergency maintenance item.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.3 Diesel Fuel Oil and Starting Air**

NRC RAI No: **3.8.3--10**

RAI STATEMENT:

--Bases for ITS SR 3.8.3.4, STS Bases markup page B 3.8-47,
--insert B3.8-47-06
--Bases for ITS 3.8.3.4

The Bases for ITS SR 3.8.3.4 states "The IP3 offsite DG fuel oil reserve is normally stored in the same tanks used to store the IP2 offsite DG fuel oil reserve."
Comment: Confirm that the volume of fuel oil that is maintained in the offsite DG fuel oil reserve is sufficient to meet the Technical Specification requirements for IP2 and IP3, that is , the volume of oil exceeds the summation of the IP2 and IP3 requirements, or provide the current licensing basis that allows sharing between the units. Additionally, the Bases for ITS 3.8.3.4 should be expanded to explain this arrangement.

NYPA RESPONSE:

CTS 3.7.A.5 specifies that: "30,026 gallons of fuel compatible for operation with the diesels shall be available onsite or at the Buchanan substation. This 30,026 gallon reserve is for Indian Point Unit No. 3 usage only and is in addition to the fuel requirements for other nuclear units on the site."

IP3 ITS 3.8.3.1 maintains the requirement, except that updated NYPA analyses support a lesser volume requirement as stated in the reply to RAI 3.8.3 - 2: "Verify reserve storage tank(s) contain (greater than or equal to) 26826 gal of fuel oil reserved for IP3 usage only."

Note that DOC M.1 changes the Frequency for verification that IP3's 30,026 gallons of fuel is maintained for IP3 from weekly to 24 hours. Therefore, ITS maintains tighter controls than CTS in ensuring that IP3's fuel will always be available for IP3.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.4 DC Sources - Operating**

NRC RAI No: **BSR 3.8.4--01**

RAI STATEMENT:

--CTS 4.6.B.3
--DOC LA.1
For Information Only:
TSTF 199 has not been accepted by the staff. Therefore, the requirement for a visual inspection of batteries is still a part of LCO 3.8.4 in NUREG-1431.
Comment: Since the CTS include a visual inspection requirement, the proposal to delete said requirement constitutes a beyond scope issue.

NYPA RESPONSE:

NYPA revised ITS to maintain the requirements of CTS 4.6.B.3 as ITS SR 3.8.4.5:
"Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration."

SR 3.8.4.5 has a 24-month Frequency consistent with CTS 4.6.B.3.

This change also results in the deletion of DOC LA.1 which proposed reloation of this requirement to the FSAR.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.4 DC Sources - Operating**

NRC RAI No: **3.8.4--01**

RAI STATEMENT:

--ITS SR 3.8.4.1 - Insert 3.8-24-04

Why are different voltages specified for batteries 31 and 32, and 33 and 34? What is the basis for these values? Do these values represent a fully charged battery?
Comment: Licensee is requested to provide detailed explanation.

NYPA RESPONSE:

See response to RAI 3.8.4-05 for NYPA proposed changes to the acceptance criteria.

Revised IP3 ITS SR 3.8.4.1 will verify a minimum of 2.13 volts per cell as a verification of a fully charged battery. Acceptance criteria for batteries 31 and 32 which have 58 cells is 123.5 V; and, acceptance criteria for batteries 33 and 34 which have 60 cells is 127.8 V.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.4 DC Sources - Operating**

NRC RAI No: **3.8.4--02**

RAI STATEMENT:

--NUREG SR 3.8.4.3

CTS includes a requirement for visual inspection. Therefore, this SR should be retained. See also item BSR 3.8.4-01 above.
Comment: Retain STS SR 3.8.4.3 requirement or request for a TS Change.

NYPA RESPONSE:

See Response to RAI BSR 3.8.4-01

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.4 DC Sources - Operating**

NRC RAI No: **3.8.4--03**

RAI STATEMENT:

--Bases Background for ITS 3.8.4, STS Bases markup page B 3.8-51, second paragraph
--Bases Background for STS SR 3.8.4

The Bases Background for STS SR 3.8.4 states, "Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours and to perform three complete cycles of intermittent loads ..." This material has not been adopted in the Bases Background for corresponding ITS SR 3.8.4.

Comment: No justification has been provided to support this proposed difference. Revise the submittal to provide the appropriate justification or to expand the Bases to address battery storage capacity.

NYPA RESPONSE:

The NUREG-1431 Bases description of the batteries was replaced with Insert: B 3.8-50-02 which is the IP3 FSAR description. Insert B 3.8-50-02 states:

Each of the four station batteries is sized to carry its expected shutdown loads for a period of 2 hours without battery terminal voltage falling below 105 volts following a plant trip that includes a loss of all AC power. Major loads with their approximate operating times on each battery are listed in Reference 4. The four battery chargers have been sized to recharge discharged batteries within 15 hours while carrying the normal DC subsystem load.

Information regarding the capability of the batteries to perform three complete cycles of intermittent loads was not found in the IP3 licensing basis and was not included in the ITS bases.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.4 DC Sources - Operating**

NRC RAI No: **3.8.4--04**

RAI STATEMENT:

--Bases Page B3.8-51 - Background

In the last paragraph of this Bases discussion, the NUREG term "fully" is changed to "required" with respect to the charged state of the batteries. What is the purpose of this change? Comment: Can this be interpreted to mean that the licensee does not consider it necessary to maintain the batteries in a fully charged condition? See also Q3.8.4-01.

NYPA RESPONSE:

NYPA believes that the term "fully charged" implies that the battery has just completed an equalizing charge and that the term "required charge" implies that the battery meets the requirements of ITS LCO 3.8.6.

NYPA revised ITS to ensure this distinction is properly understood. The revised wording is underlined in the following excerpt from the Bases:

"Each DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery <u>charged as necessary to meet the requirements of LCO 3.8.6, Battery Parameters.</u> Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to the required charged state within 15 hours while supplying normal steady state loads discussed in the FSAR, Chapter 8 (Ref. 4)."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.4 DC Sources - Operating**

NRC RAI No: **3.8.4--05**

RAI STATEMENT:

--Bases Page B3.8-54 - SR 3.8.4.1

This SR and associated Bases discussion are intended to address the float condition of the battery. The voltage of each cell (for a typical rectangular cell battery) is 2.13V or higher. Comment: The value of 2.07 Volts per cell proposed for inclusion in this Bases discussion appears to be the open circuit voltage of each cell, not the float voltage. It appears that some correction is required.

NYPA RESPONSE:

NYPA agrees with this comment. CTS 4.6.B.1 and associated implementing procedure verify float voltage of at least 2.13 volts or higher. NYPA revised ITS so that IP3 ITS SR 3.8.4.1 verifies a minimum of 2.13 volts per cell as follows:

Verify battery terminal voltage on float charge is within the following limits:

a. 123.5 V for batteries 31 and 32; and

b. 127.8 V for batteries 33 and 34.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.4 DC Sources - Operating**

NRC RAI No: **3.8.4--06**

RAI STATEMENT:

--Bases Page B3.8-55 - NUREG SR 3.8.4.3

This requirement for visual inspection of the batteries should be retained. TSTF 199 has not been accepted by the staff. Therefore, the requirement for a visual inspection of batteries is still a part of LCO 3.8.4 in NUREG-1431.
Comment: Since the CTS include a visual inspection requirement, the proposal to delete the said requirement constitutes a beyond scope issue if not retained (See BSR Q3.8.4-01.)

NYPA RESPONSE:

See Response to RAI BSR 3.8.4-01

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.4 DC Sources - Operating**

NRC RAI No: **3.8.4--07**

RAI STATEMENT:

DOC M.2
ITS SR 3.8.4.2
STS SR 3.8.4.6
CTS 4.6.B
Bases for ITS SR 3.8.4.2, STS Bases markup page B 3.8-56,
inserts B3.8-51-01 and B3.8-51-02

The licensee is requested to provide a justification for why the battery charger surveillance requirement acceptance criteria (stated in terms of amps and hours of operation) are not included in ITS 3.8.4.2. DOC M2 does not provide this information.
Comment: Revise ITS SR 3.8.4.2 to provide the specific acceptance criteria for each battery charger in order to conform to the STS format for this Surveillance Requirement.

NYPA RESPONSE:

FSAR 8.2 states: "The four battery chargers have been sized to recharge the above partially discharged batteries within 15 hours while carrying its normal load." However, there is no CTS or FSAR requirement for the periodic re-verification that this requirement is met. IP3 is voluntarily adopting a requirement for periodic re-verification of battery charger capacity per DOC M.2. Procedures and acceptance criteria for this new requirement are under development.

As stated in IP3 ITS 3.8.4, DOC M.2, the specific acceptance criteria for each battery charger will be identified in the FSAR. Not including the acceptance criteria for battery charger capacity in the ITS is consistent with NUREG-1431's treatment of the acceptance criteria for battery capacity in SR 3.8.4.7 (IP3 ITS 3.8.4.3).

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.5 DC Sources - Shutdown**

NRC RAI No: **3.8.5--01**

RAI STATEMENT:

--Bases Page B3.8-61 - Insert B3.8-61-01

What is the justification for the proposal to include this insert in the Bases? What cross connects are envisioned with the proposed Bases addition, and how are they made? Comment: Provide justification and functional description in details.

NYPA RESPONSE:

The revision 0 version of Insert B3.8-61-01 states:

DC subsystems may be cross connected in Modes 5 and 6 and during movement of irradiated fuel because there is no requirement to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem.

NYPA revised ITS to expand the insert as follows:

DC subsystems 31 and 32 may be cross connected and powered by battery 31 or 32, and both DC subsystems remain OPERABLE (Ref. 2). Similarly, DC subsystems 33 and 34 may be cross connected and powered by battery 33 or 34. However, only one pair of subsystems at a time may be cross connected. Cross connecting DC subsystems in Modes 5 and 6 and during movement of irradiated fuel is acceptable because there is no requirement for redundancy or separation between DC busses when the plant is in this condition. Both DC subsystems in the cross connected pair remain OPERABLE even when powered by one battery because the capacity of one battery is adequate to carry the loads on both busses when the plant is in this condition.

Revise this insert to read as follows: DC subsystems may be cross connected in Modes 5 and 6 and during movement of irradiated fuel because there is no requirement for redundancy or separation between DC busses when the plant is in this condition.

NYPA included this clarification in the Bases to support use of this plant specific design feature as described in FSAR Chapter 8.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.5 DC Sources - Shutdown**

NRC RAI No: **3.8.5--02**

RAI STATEMENT:

--Bases Page B3.8-61- Actions A.1, A.2.1, etc.

The first sentence of this Bases discussion is proposed to be modified to state "If any DC electrical subsystems are required by LCO 3.8.10,..." what is the intent of this proposed change? Are there plant conditions during which the licensee feels that no DC subsystems will be required OPERABLE? If so, what are they and why would DC power not be required? Comment: This Bases discussion may require revision depending on the licensee's response.

NYPA RESPONSE:

NYPA revised ITS so that the Bases for IP3 ITS 3.8.5, Required Actions, read as follows:

If any DC electrical subsystem required by LCO 3.8.10 becomes inoperable, the remaining DC power available may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.8 Inverters - Shutdown**

NRC RAI No: **3.8.7 / .8--01**

RAI STATEMENT:

--ITS SR 3.8.7.1
--STS SR 3.8.7.1

STS 3.8.7.1 requires verifying correct inverter frequency on a plant specific basis. This requirement has not been adopted in corresponding ITS SR 3.8.7.1.
Comment: No justification has been provided to support this proposed difference. Revise the submittal to explain why this requirement is not applicable, or conform to the STS.

NYPA RESPONSE:

NYPA did not include the requirement for periodic verification of inverter frequency because only 3 of the 4 inverters has the instrumentation required to perform this SR.

NYPA revised ITS so that SR 3.8.7.1 and SR 3.8.8.1 include the requirement for periodic verification of inverter frequency as specified in NUREG-1431. SR 3.8.7.1 and SR 3.8.8.1 is now modified by a Note that states: "Frequency verification not required to be performed for inverter 34." This difference from NUREG-1431 is justified using JFD DB.1. The Bases associated with SR 3.8.7.1 and SR 3.8.8.1 was also revised to explain that inverter frequency indication is not available for inverter 34.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.8 Inverters - Shutdown**

NRC RAI No: **3.8.8--01**

RAI STATEMENT:

--Bases Page B3.8-76 - Insert B3.8-76-01

This insert addresses cross connecting electrical buses (presumably DC buses). This Bases discussion may have to be revised depending on the licensee response to Q3.8.5-01.

NYPA RESPONSE:

INSERT B 3.8-76-01 states: This LCO does not require OPERABILITY of the constant voltage transformers (CVTs) capable of supplying VIB 34 even if inverter 34 is required to be OPERABLE. This is acceptable because VIB 34 will be powered from battery 34 via inverter 34 for a minimum of 2 hours and electrical buses may be cross connected as needed to support inverter 34 prior to the depletion of battery 34.

NYPA believes that this configuration is acceptable because there is no requirement for redundancy or separation between DC busses when the plant is in Modes 5 and 6.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **3.8.9 Distribution Systems - Operating**

NRC RAI No: **3.8.9--01**

RAI STATEMENT:

--LCO 3.8.9, Condition B
--Bases Page B3.8-79, Insert B3.8.79-03
--Bases Table B3.8.9-1

LCO 3.8.9, Condition B addresses one vital instrument bus inoperable, and Insert B3.8-79-03 states that there are four vital instrument buses. However, Table B3.8.9-1 lists a total of 8 vital instrument buses. Which is correct? Insert B3.8-79-03 which states there are 4 buses, or Table B3.8.9-1 which lists 8 buses? When is Condition B of LCO 3.8.9 invoked?
Comment: The licensee should provide a response to the staff's questions and provide Bases revisions, as necessary, to eliminate any confusion. Note that correction to Bases discussions other than 3.8.9 may also be required.

NYPA RESPONSE:

There are 4 vital instrument buses, each consisting of two interconnected buses. For example, interconnected buses 31 and 31A constitute a single vital instrument bus for purposes of applying LCO 3.8.9, Condition B. NYPA revised the Bases to clarify this arrangement.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **4.0 DESIGN FEATURES**

NRC RAI No: **4.0--01**

RAI STATEMENT:

DOC A.8 states that CTS 5.4 Fuel Storage does not specify any requirements for the fuel racks or the spent fuel storage facility or the design limitations.

ITS 4.3 Fuel Storage adds ITS4.3.1.1.c, ITS4.3.1.1d, ITS 4.3.1.2.c, and ITS4.3.2.
Comments: DOC A.8 should be changed to a more restrictive TS change, not merely an administrative change, because these changes would affect safety. From the perspective of TS, these are more restrictive.

NYPA RESPONSE:

NYPA revised ITS so that information being moved into Technical Specifications in ITS4.3.1.1.c, ITS4.3.1.1d, ITS 4.3.1.2.c, and ITS4.3.2 is identified as more restrictive change M.1 instead of administrative change A.8.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **4.0 DESIGN FEATURES**

NRC RAI No: **4.0--02**

RAI STATEMENT:

ITS 4.3.1.1b deleted "if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]" from the STS 4.3.1.1b, and added Insert 4.0-1-03.

Comments: Provide JDC for this change with technical reasons.

NYPA RESPONSE:

NYPA revised ITS so that ITS 4.3.1.1.b reads as follows: b. keff {less than or equal to} 0.95 if the assemblies are inserted in accordance with Technical Specification 3.7.16, Spent Fuel Assembly Storage. In conjunction with this change, ITS 4.0, DOC A.7, is deleted.

This change maintains CLB consistent with CTS 5.4.1.2. JFD CLB.3 is added to explain that the change to the STS maintains CLB.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **4.0 DESIGN FEATURES**

NRC RAI No: **4.0--03**

RAI STATEMENT:

ITS 4.3.1.2b deleted "if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR] from the STS 4.3.1.2b, and added Insert 4.0-2-01. Also, 4.3.1.2 deleted the STS4.3.1.2c without marked JDC.

Comments: Provide JDC for these changes with technical reasons.

NYPA RESPONSE:

IP3 ITS 4.3.1.2.b requirements for reactivity limits for new fuel storage racks replace the term "if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR]" with the term "under all possible moderation conditions. Credit may be taken for burnable integral neutron absorbers." This is a verbatim transcription of CTS 5.4.2 and maintains CLB.

As stated in ITS 4.0, JFD CLB.1, NUREG-1431, Section 4.3.1.2.c requirement that keff be $\leq$ 0.98 if the new fuel storage racks are moderated with aqueous foam is not included in the ITS. The new fuel storage racks are designed to ensure keff $\leq$ 0.95 under all possible moderation conditions. This change maintains the current licensing basis.

NYPA revised ITS so that changes to NUREG-1431, Section 4.3.1.2, are marked as CLB.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **5.5.1 Offsite Dose Calculation Manual (ODCM)**

NRC RAI No: **5.5--01**

RAI STATEMENT:

--STS 5.5.6
--JFD None

STS Section 5.5.6 "Pre-Stressed Concrete Containment Tendon Surveillance Program" has been removed, deleted, or relocated without any discussion or justification.

Comment: Revise the submittal to either include STS Section 5.5.6 "Pre-Stressed Concrete Containment Tendon Surveillance Program" or provide justification for not including it in the ITS.

NYPA RESPONSE:

IP3 design does not include "Pre-Stressed Concrete Containment Tendons."

NYPA revised ITS so that ITS 5.5.5, Component Cyclic or Transient Limit, includes JFD DB.1 that states that STS 5.5.6, Pre-Stressed Concrete Containment Tendon Surveillance Program, is not in ITS because IP3 design does not include Pre-Stressed Concrete Containment Tendons.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **5.5.8 Steam Generator (SG) Tube Surveillance Program**

NRC RAI No: **5.5--02**

RAI STATEMENT:

--CTS 4.9.C.3/3.1.F.7
--DOC A.7/A.8
--ITS 5.5.8

Several reports in CTS Section 4.9.C.3/3.1.F.7 (associated with ITS 5.5.8) have been deleted. DOC A.7 and A.8 essentially state that these reports are required by 10 CFR 50.73 and that the changes are Administrative in nature. 10 CFR 50.73 does not specifically state the requirement of these reports. Because the deletion of these two details constitutes less specific requirements, this would appear to be a "Less Restrictive" type change.

Comment: Revise the submittal for DOCs A.7 and A.8 to address how 10 CFR 50.73 will require these specific reports and/or re-classify these A-DOCs to L-DOCs..

NYPA RESPONSE:

DOC A.7 states that CTS 4.9.C.3 requires NRC notification within 15 days if results of SG tube inspections fall into Category C-3 (i.e., more than 10% of total tubes inspected are degraded or more than 1% of tubes inspected are defective). ITS 5.5.8.e.3 maintains the requirement for a report within 15 days. The detail that ITS does not retain is that, "The written follow-up of this report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence." Category C-3 results for a SG inspection may meet one or more situations covered by 10 CFR 50.73 (e.g., a condition of the nuclear power plant, including its principal safety barriers, being seriously degraded). Therefore, CTS 4.9.C.3 and ITS 5.5.8.e.3 serve no purpose other than to accelerate the requirement for the initial report. NYPA revised ITS to provide more detail in ITS 5.5.8, DOC A.7.

DOC A.8 deletes CTS 3.1.F.7 which requires that NYPA must inform the NRC before the reactor is brought critical after the reactor is shut down, or a steam generator removed from service, to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube. This NYPA specific requirement was established before IP3 replaced SGs to correct SG tube leakage problems. NYPA revised ITS to delete this requirement as DOC L.1.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **5.5.9 Secondary Water Chemistry Program**

NRC RAI No: **5.5--03**

RAI STATEMENT:

--STS 5.5.10
--ITS 5.5.9
--JFD None

STS Section 5.5.10 "Secondary Water Chemistry Program" has been changed without any discussion or justification. The term "discharge of condensate pumps" has been deleted and replaced by the term "condensate hot wells." Also the term "and low pressure turbine disc stress corrosion cracking" has been deleted. Comment: Revise the submittal to either include original STS Section 5.5.10 "Secondary Water Chemistry Program" wording or provide justification for this change.

NYPA RESPONSE:

NYPA revised ITS to notate that deletion of the term "and low pressure turbine disc stress corrosion cracking" included in the STS was not included in the IP3 ITS consistent with current licensing basis. Justification is provided in FSAR Appendix 14A, Likelihood and Consequences of Turbine Overspeed at Indian Point 3.

NYPA revised ITS to notate that the term "discharge of condensate pumps" was deleted and replaced by the term "condensate hot wells" consistent with current licensing basis.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **5.5.10 Ventilation Filter Testing Program (VFTP)**

NRC RAI No: **5.5--04**

RAI STATEMENT:

--CTS 4.5.A.4.b/4.5.A.4.c/4.5.A.5.c/4.5.A.5.d
--ITS 5.5.10
CTS 4.5.A.4.b/4.5.A.4.c/4.5.A.5.c/4.5.A.5.d use the term "at least once per 24 months..." where as the corresponding ITS 5.5.10 states "after 24 months of standby service..." These two terms have different meanings. There is no DOC or JFD for these changes. Comment: Revise the submittal to either include original STS Section 5.5.10 "Ventilation Filter Testing Program (VFTP)" wording or provide justification (or DOC) for this change.

NYPA RESPONSE:

NYPA revised ITS so that ITS 5.5.10 2 reads:
"Every 24 months," instead of, "After 24 months of standby service."

NYPA revised ITS so that ITS 5.5.10 3 reads:
"Every 18 months," instead of, "After 18 months of standby service."

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **5.5.12 Diesel Fuel Oil Testing Program**

NRC RAI No: **5.5--05**

RAI STATEMENT:

--ITS 5.5.12 - insert 5.0-15-01
--JFD CLB-1

The proposed change (insert 5.0-15-01) indicates that the fuel oil program will only be applicable to the fuel oil stored in or added to the storage tanks. Comment: Why is it acceptable for the program to be not applicable to the reserve fuel oil tanks?

NYPA RESPONSE:

NYPA revised ITS so that Diesel Fuel Oil Testing Program requirements generally conform to program requirements specified in NUREG-1431 including TSTF-106 and 118. The program will include testing requirements which are applicable to both the onsite DG fuel oil tanks and the reserve fuel oil tanks. Neither the CTS or the FSAR establish any requirements for diesel fuel oil testing programs or acceptance criteria for fuel oil parameters. Deviations between NUREG-1431 and the IP3 ITS will reflect plant specific design for the storage tanks and maintains or improves the current practice for assuring the quality of new and stored fuel oil.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **5.5.12 Diesel Fuel Oil Testing Program**

NRC RAI No: **5.5--06**

RAI STATEMENT:

--ITS 5.5.12 - insert 5.0-16-01
--JFD CLB-2

For the proposed change (insert 5.0-16-01) it is not clear to the NRC staff what the significance of the proposed change is. It is the staff's belief that the manufacturer of the EDG at IP3 recommends #2 diesel fuel for the use in the engines. Basically the only ASTM standard that addresses #2 diesel fuel (ASTM2D) is D-975. Comment: Given this, what is the purpose of deleting the direct reference to D-975 in favor of an indirect route which ultimately appears that it will lead to the same reference.

NYPA RESPONSE:

NYPA revised ITS so that Diesel Fuel Oil Testing Program requirements conform to requirements in NUREG-1431 including TSTF-106 and 118. See response to RAI 5.5-05.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **5.5.12 Diesel Fuel Oil Testing Program**

NRC RAI No: **5.5--07**

RAI STATEMENT:

--ITS 5.5.12 - insert 5.0-16-02
--JFD CLB-2

For the proposed change (insert 5.0-16-02) the term "commercial grade diesel fuel" is used. Comment: What is "commercial grade diesel fuel and how does it differ from ASTM2D fuel? What is the basis for stating that this "commercial grade" fuel oil is compatible in the IP3 EDG?

NYPA RESPONSE:

NYPA revised ITS so that Diesel Fuel Oil Testing Program requirements conform to requirements in NUREG-1431 including TSTF-106 and 118. See response to RAI 5.5-05. The program is revised to delete reference to 'commercial grade diesel fuel' and applies requirements consistent with ASTM standards.

NYPA REPLY TO NRC RAI
REGARDING REVISION 0 OF PROPOSED ITS

ITS LCO: **5.5.12 Diesel Fuel Oil Testing Program**

NRC RAI No: **5.5--08**

RAI STATEMENT:

--ITS 5.5.12.c
--JFD CLB-2

For the proposed change (ITS 5.5.12.c) the term "ASTM D-2276" has been deleted. This deletion may lead to confusion because the purposed substitution (applicable ASTM standards) is not specific enough. Comment: ASTM D-2276 is a particulate test for aviation fuel that has been adopted for use with diesel fuel. However, if a search of the ASTM standard was conducted to find a particulate test for diesel fuel, none would be found. This reference to ASTM D-2276 should be retained in the ITS

NYPA RESPONSE:

NYPA revised ITS so that Diesel Fuel Oil Testing Program requirements apply to both the onsite DG fuel oil storage tanks and the offsite DG reserve fuel oil storage tanks. Additionally, ITS 5.5.12 is revised to conform to requirements in NUREG-1431 including TSTF-106 and 118. The program will reference ASTM D-2276 as the applicable test method for measuring particulate concentration in fuel oil.

ATTACHMENT III TO IPN-00-069

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 3.4.1:
"RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average loop temperature limit is consistent with full power operation within the nominal operational envelope. RCS average loop temperature is assumed to be the highest indicated value of the Tavg indicators and this is the value that is compared to the acceptance criteria. A higher average temperature will cause the core to approach DNB limits. RAI-3

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. RCS flow rate is determined by calculating the average flow rate for each loop and then calculating the sum of these average loop flow rates and this sum of the averages is compared to the acceptance criteria. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

state condition following load changes and other expected transient operations. Pressurizer pressure indications are averaged to determine the value for comparison to the LCO limit. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average loop temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. RCS average loop temperature is assumed to be the highest indicated value of the Tavg indicators and this is the value that is compared to the acceptance criteria. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

RAI-3

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 24 months verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

(continued)

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.1:
"RCS DNB 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
3.1-36	175	191	Deleted footnote regarding applicability of DNB analysis margin for Cycle 10
3.1-37	175	175	
3.1-38	175	175	
3.1-39	170	170	
T4.1-1(1)	170;98-043	185	No impact
T4.1-1(6)	181;98-043	185	No impact
4.3-4	175	175	

A.1 A.2

~~3.1~~ Reactor Coolant System (RCS)

LCO 3.4.1 ~~H.~~ RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits*

Specification

LCO 3.4.1 ~~1.~~ During ~~the POWER OPERATION CONDITION~~ Mode 1 (A.1), RCS DNB parameters for pressurizer pressure and RCS average temperature shall be within the limits specified below:

LCO 3.4.1.a ~~a.~~ Pressurizer pressure ≥ 2205 psig; (A.7)

LCO 3.4.1.b ~~b.~~ Maximum ~~indicated~~ T_{avg} ≤ 571.5°F; and

~~2.~~ ~~At the POWER OPERATION CONDITION~~ Mode 1 (A.1) with ~~four reactor coolant pumps running~~ (A.3), the RCS DNB parameter for RCS total flow rate shall be within the following limit:

LCO 3.4.1.c RCS total flow rate ≥ 375,600 gpm.

3.4.1 Applicability Note ~~3.~~ The pressurizer pressure limit of Specification ~~3.1.H.1~~ does not apply during:

~~a.~~ THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

3.4.1, Req Act A.1 ~~4.~~ If pressurizer pressure, RCS average temperature, or RCS total flow rate are not in accordance with Specifications 3.1.H.1, 3.1.H.2, or 3.1.H.3, then, ~~immediately verify that the safety limits of Specification 2.1 have not been exceeded and~~ (A.4) within 2 hours, restore the RCS DNB parameter(s) to within limits.

Req Act B.1 ~~5.~~ If pressurizer pressure and/or RCS average temperature are not restored to within limits within 2 hours, be in the HOT SHUTDOWN CONDITION Mode 2 (L.1) within 6 hours.

Req Act B.1 ~~6.~~ If RCS total flow rate is not restored to within the limits of Specification 3.1.H.2 within 2 hours, bring THERMAL POWER to ≤ 10% RTP within 6 hours ~~and ensure operation is in accordance with Specification 3.1.A.1.e.~~ (M.1)

Surveillance Requirements

~~Reference Technical Specification Table 4.1-1, Items 4, 5, and 7, and Section 4.3.B.~~ (A.1)

~~Bases~~

~~**Background**~~
~~These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS~~ (A.1)

~~* Current DNB analysis contains adequate margin for Cycle 10. Prior to achieving criticality in Cycle 11, the DNB analysis must be reviewed and approved by NRC staff.~~ R.1

Deleted by CTS Amendment 191 (See next page)

Amendment No. ~~170~~, 175

3.1 Reactor Coolant System (RCS)

H. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

Specification

1. During the POWER OPERATION CONDITION, RCS DNB parameters for pressurizer pressure and RCS average temperature shall be within the limits specified below:

   a. Pressurizer pressure ≥ 2205 psig;

   b. Maximum indicated T_{avg} ≤ 571.5°F; and

2. At the POWER OPERATION CONDITION with four reactor coolant pumps running, the RCS DNB parameter for RCS total flow rate shall be within the following limit:

   RCS total flow rate ≥ 375,600 gpm.

3. The pressurizer pressure limit of Specification 3.1.H.1 does not apply during:

   a. THERMAL POWER ramp > 5% RTP per minute; or

   b. THERMAL POWER step > 10% RTP.

4. If pressurizer pressure, RCS average temperature, or RCS total flow rate are not in accordance with Specifications 3.1.H.1, 3.1.H.2, or 3.1.H.3, then, immediately verify that the safety limits of Specification 2.1 have not been exceeded and, within 2 hours, restore the RCS DNB parameter(s) to within limits.

5. If pressurizer pressure and/or RCS average temperature are not restored to within limits within 2 hours, be in the HOT SHUTDOWN CONDITION within 6 hours.

6. If RCS total flow rate is not restored to within the limits of Specification 3.1.H.2 within 2 hours, bring THERMAL POWER to ≤ 10% RTP within 6 hours and ensure operation is in accordance with Specification 3.1.A.1.e.

Surveillance Requirements

Reference Technical Specification Table 4.1-1, Items 4, 5, and 7, and Section 4.3.B.

Bases

Background

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.1:

"RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

Additionally, achieving the specified flow rate requires four RCPs in operation. This is an administrative change with no adverse impact on safety.

A.4 CTS 3.1.H.4 specifies that if the RCS pressure, temperature or flow limits of CTS 3.1.H are exceeded, then the safety limits of specification 2.1 must be verified. ITS 3.4.1, Required Actions, do not specify this requirement. Not including a specific requirement to verify SLs are met when LCO 3.4.1 limits are not met is acceptable because ITS SL 2.1.1 already specifies Actions if SLs are violated (i.e., restore compliance and be in Mode 3 within 1 hour). Additionally, ITS 3.4.1 Bases specify that safety limits for DNB related parameters are provided in ITS SL 2.1.1 and that the operator must check whether or not an SL may have been exceeded if LCO 3.4.1 limits are not met. Therefore, this is an administrative change with no impact on safety.

A.5 CTS 4.3.B requires verification by "flow calculation" every 24 months that RCS total flow rate is within required limits. ITS SR 3.4.1.4 maintains this requirement except that the ITS specifies use of a precision calorimetric heat balance. This is an administrative change with no adverse impact on safety because a precision calorimetric heat balance is a specific description of the intent of the flow calculation required by CTS 4.3.B.

A.6 Superceded by CTS Amendment 191.

A.7 CTS 3.1.H.1.b specifies a limit on the "maximum indicated" Tavg. ITS LCO 3.4.1.b and ITS SR 3.4.1.2 maintain this limit on the reactor coolant system average temperature with a clarification in the ITS Bases that RCS average loop temperature is assumed to be the highest indicated value of the Tavg indicators and this is the value that is compared to the acceptance criteria. This is an administrative change with no impact on safety because the combination of the ITS LCO 3.4.1.b and ITS SR 3.4.1.2 requirements with the Bases clarification provides a more definitive description of the existing CTS requirement.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.1:
"RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

loop

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. ~~Indications of temperature are averaged to determine a value for comparison to the limit.~~ A higher average temperature will cause the core to approach DNB limits. PA.1

Insert: B 3.4-1-01

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. ~~Flow rate indications are averaged to come up with a value for comparison to the limit.~~ A lower RCS flow will cause the core to approach DNB limits.

Insert: B 3.4-1-02

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event. DB.1

Insert: B 3.4-1-c

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.1 - RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

INSERT: B 3.4-1-01

PA.1

RCS average loop temperature is assumed to be the highest indicated value of the Tavg indicators and this is the value that is compared to the acceptance criteria.

R.1

INSERT: B 3.4-1-02

PA.1

RCS flow rate is determined by calculating the average flow rate for each loop and then calculating the sum of these average loop flow rates and this sum of the averages is compared to the acceptance criteria.

INSERT: B 3.4-1-03

DR.1

Calculations have shown that reactor heat equivalent to 10% rated power can be removed via the steam generators with natural circulation without violating DNBR limits. This analysis assumed conservative flow resistance including steam generator tube plugging and a locked rotor in each loop (Ref. 1).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

following load changes and other expected transient operations. Insert: B3.4-5-01 The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every ~~[18]~~ 24 months Insert: B3.4-5-02 ~~allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate~~.

The Frequency of ~~[18]~~ 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered, SG Tubes plugged, or other activities performed, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ [90%] RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of [90%] RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching [90%] RTP.

REFERENCES

1. FSAR, Section ~~[15]~~ 14.

INSERT: B 3.4-5-01

PA 1

RCS average loop temperature is assumed to be the highest indicated value of the Tavg indicators and this is the value that is compared to the acceptance criteria.

INSERT: B 3.4-5-02

PA 1

verifies that the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.3:
"RCS Pressure and Temperature (P/T) Limits"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in Figure 3.4.3-1, Figure 3.4.3-2, and Figure 3.4.3-3. RAI-5

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE-------- Required Action A.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure < 500 psig.	6 hours 36 hours

RAI-10

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed whenever this Condition is entered. -------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	<u>AND</u>		
	C.2	Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	------------------NOTE------------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the following: a. Figure 3.4.3-1 during RCS heatup; b. Figure 3.4.3-2 during RCS cooldown; and c. Figure 3.4.3-3 during RCS inservice leak and hydrostatic testing.	30 minutes

RAI-5

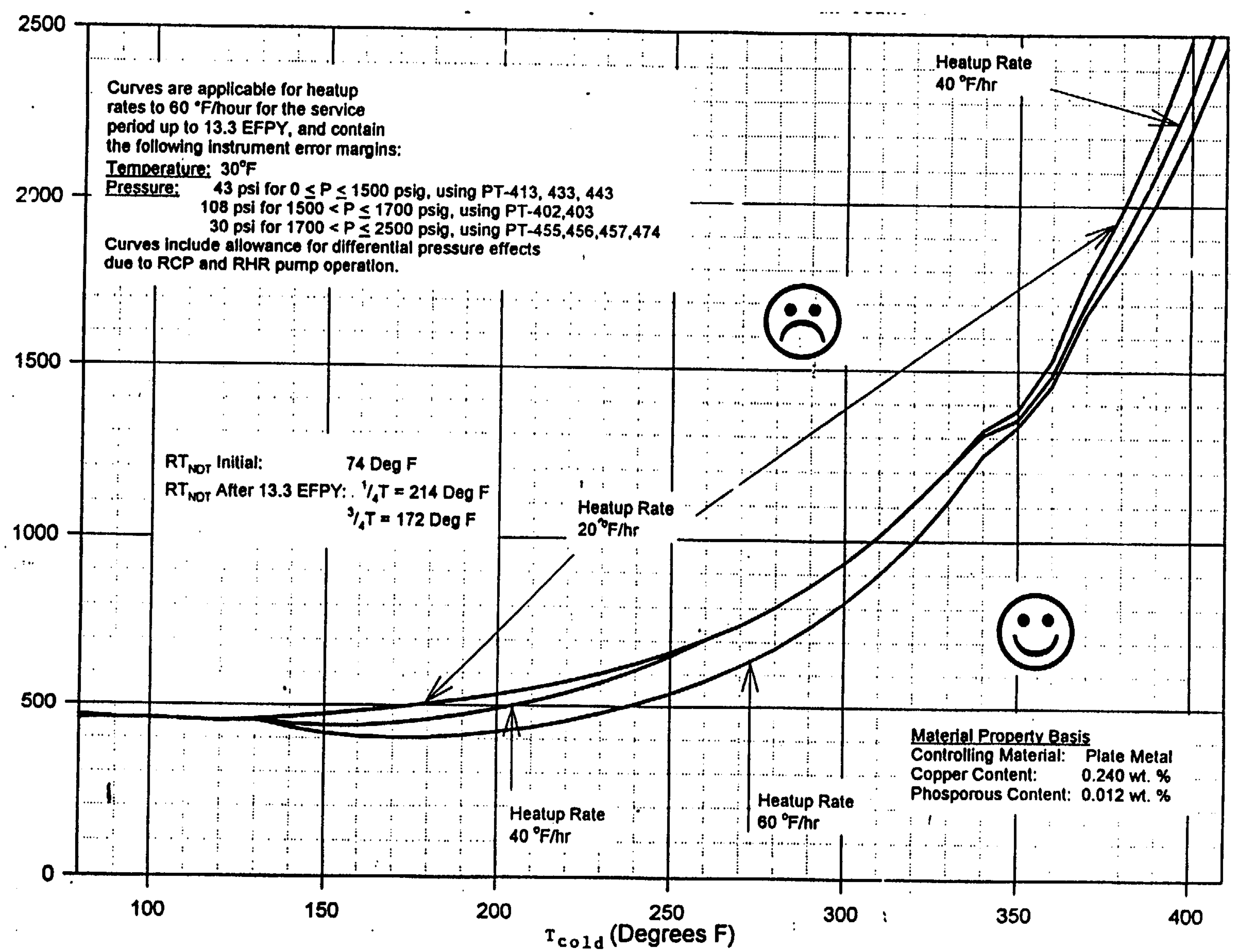
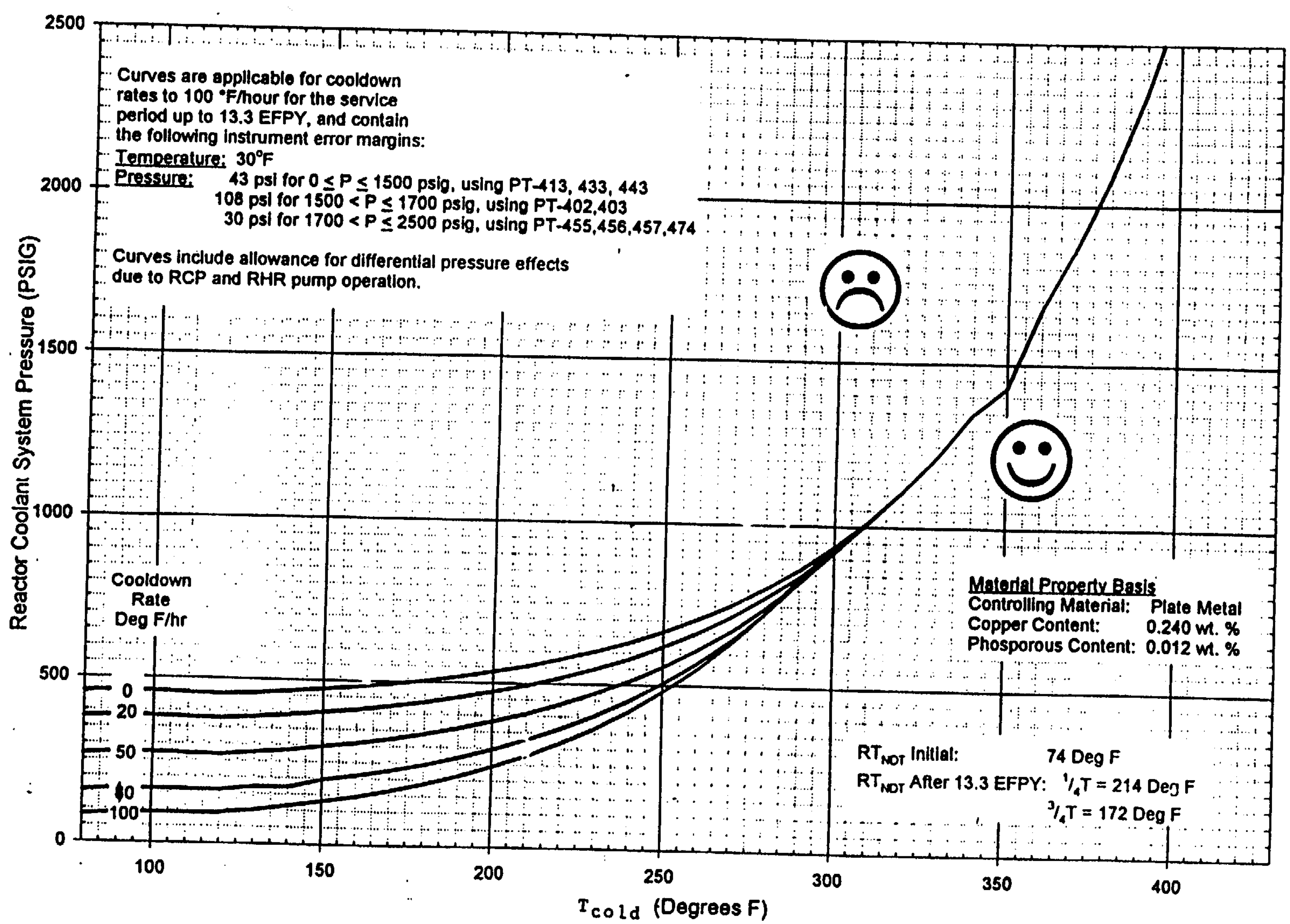


Figure 3.4.3-1:
Heatup Limitations for the Reactor Coolant System

RAI-6

Figure 3.4.3-2:
Cooldown Limitations for the Reactor Coolant System



RAI-6

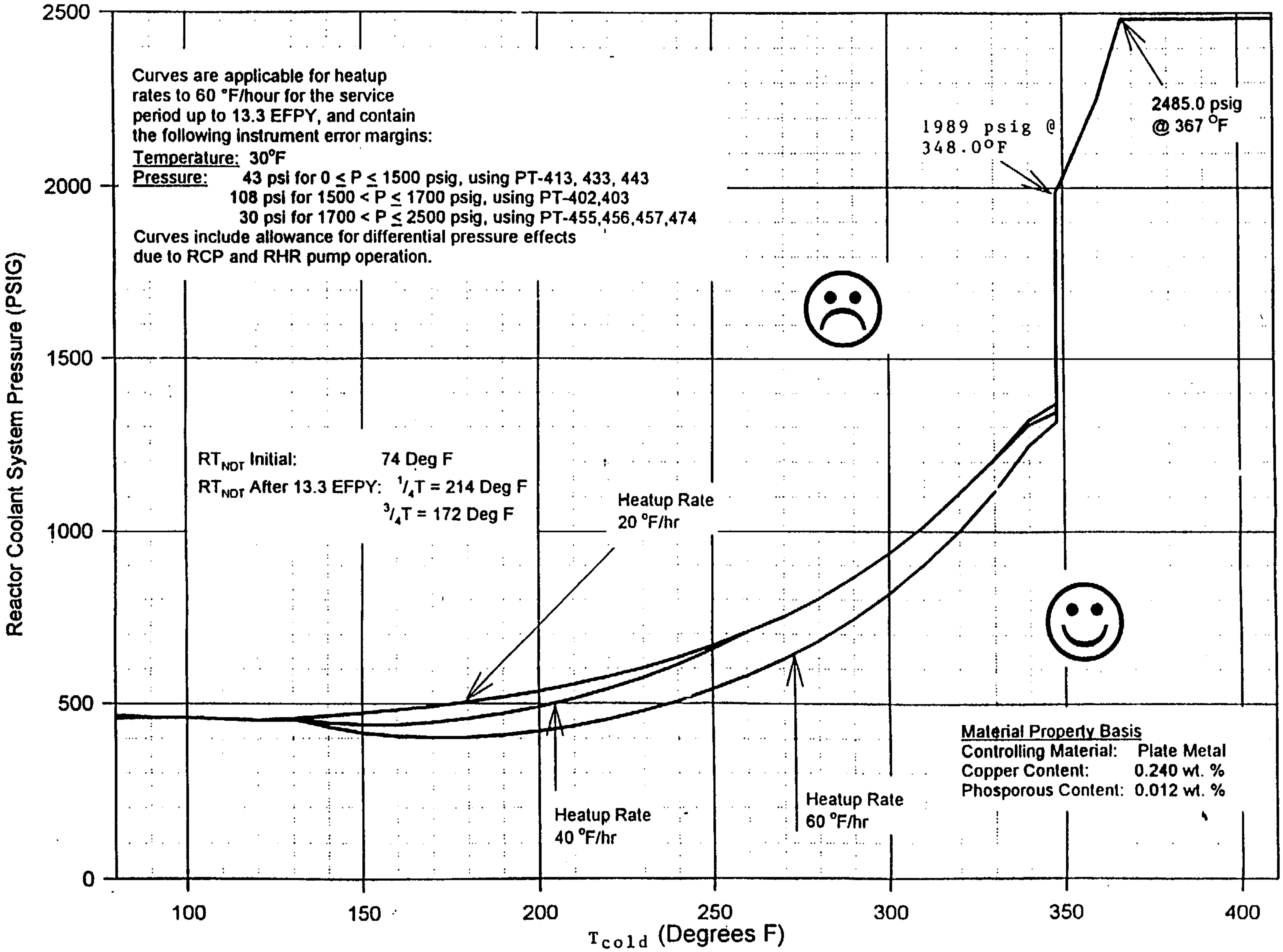


Figure 3.4.3-3:
Hydrostatic and Inservice Leak Testing Limitations for the Reactor Coolant System

RAI 6

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

LCO 3.4.3, Figure 3.4.3-1, Heatup Limitations for the Reactor Coolant System, Figure, 3.4.3-2, Cooldown Limitations for the Reactor Coolant System, and Figure 3.4.3-3, Hydrostatic and Inservice Leak Testing Limitations for the Reactor Coolant System, contain P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, respectively (Ref. 1). RAI-5

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. The happy face icon shown on Figure 3.4.3-1, Figure, 3.4.3-2, and Figure 3.4.3-3, indicates the side of the curve in which operation is permissible. Conversely, the sad face icon indicates the side of the curve in which operation is prohibited. NYPA

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

(continued)

BASES

BACKGROUND (continued)

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

(continued)

BASES

BACKGROUND (continued)

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36.

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

Figure 3.4.3-1, Heatup Limitations for the Reactor Coolant System, Figure, 3.4.3-2, Cooldown Limitations for the Reactor Coolant System, and Figure 3.4.3-3, Hydrostatic and Inservice Leak Testing Limitations for the Reactor Coolant System, contain P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, respectively. These figures specify the maximum RCS pressure for various heatup and cooldown rates at NYPA

(continued)

BASES

LCO (continued)

any given reactor coolant temperature. The figures provide the limiting RCS pressure and reactor coolant temperature combination for reactor coolant temperature heatup rates up to 60°F/hr and reactor coolant temperature cooldown rates up to 100°F/hr. Therefore, heatup rates that exceed 60°F/hr and cooldown rates that exceed 100°F/hr are considered not within the limits of this LCO. NYPA

The LCO limits apply to all components of the RCS pressure boundary, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves. Heatup and cooldown limits are specified in hourly increments (i.e., the heatup and cooldown limits are based on the temperature change averaged over a one hour period). Limit lines for cooldown rates between those presented may be obtained by interpolation. NYPA

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existence, size, and orientation of flaws in the vessel material. NYPA

(continued)

BASES (continued)

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours. Note that LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), will also apply and may require limits for operation that are more restrictive than or supplement this limit. NYPA

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Heatup and cooldown limits are specified in hourly increments (i.e., the heatup and cooldown limits are based on the temperature change averaged over a one hour period). Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

(continued)

BASES (continued)

REFERENCES

1. WCAP-7924-A, July 1972.

2. 10 CFR 50, Appendix G.

3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

4. ASTM E 185-70.

5. 10 CFR 50, Appendix H.

6. Regulatory Guide 1.99, Revision 2, May 1988.

7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.3:
"RCS Pressure and Temperature (P/T) Limits"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

M.2 CTS 3.1.B, Heatup and Cooldown, does not specify any required surveillances for the periodic or systematic verification that RCS pressure and temperature and RCS heatup and cooldown rates are within the specified limits. ITS SR 3.4.3.1 is added to require verification that operation is within the limits of Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes and during inservice leak and hydrostatic testing.

ITS SR 3.4.3.1 is modified by a Note that requires this SR to be performed only during system heatup, cooldown, and leak testing. Periodic verification that RCS pressure and temperature limits are met is not required in Modes 1 and 2 because LCO 3.4.2 contains a more restrictive requirements for pressure and temperature.

During those periods when ITS SR 3.4.3.1 must be performed, a Frequency of once per 30 minutes is specified because heatup and cooldown limits are specified in hourly increments (i.e., the heatup and cooldown limits are based on the change during an hour period and is consistent with CTS requirements). Therefore, formal verification at 30 minute intervals permits assessment and correction for minor deviations within a reasonable time.

These more restrictive changes are acceptable because they do not introduce any operation that is un-analyzed while requiring a more conservative response than is currently required for the verification that pressure-temperature limits are met. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

None

REMOVED DETAIL

LA.1 CTS 3.1.B, Heatup and Cooldown, and CTS 4.3, RCS Integrity Testing, include information such as the following: the information that limits must be periodically recalculated; the clarification that heatup and cooldown rates are based on the average temperature over a one hour period; and, requirements for vessel specimen removal. These details are not retained in the ITS and are relocated to the Bases for LCO 3.4.3, This change is acceptable because ITS LCO 3.4.3 maintains the requirement to meet these pressure and temperature limits.

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 3.4.3:
"RCS Pressure and Temperature (P/T) Limits"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

⟨CTS⟩

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

⟨3.1.B.1⟩

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in ~~the PTLR~~ Insert 3.4-5-01 | R.1

⟨DOC A.3⟩

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨DOC M.1⟩ A. ----------NOTE---------- Required Action A.2 shall be completed whenever this Condition is entered. ---------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
⟨DOC M.1⟩ B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure < ~~[~~500~~]~~ psig.	6 hours 36 hours | R.1

(continued)

INSERT: 3.4-5-01

Figure 3.4.3-1, Figure 3.4.3-2, and Figure 3.4.3-3. R.1

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

<DOC M.1>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

<DOC M.2>

Insert: 3.4-6-01

R.1

Insert Figures 3.4.3-1, 3.4.3-2 and 3.4.3-3

R.1

INSERT: 3.4-6-01

following:

a. Figure 3.4.3-1 during RCS heatup;

b. Figure 3.4.3-2 during RCS cooldown; and

c. Figure 3.4.3-3 during RCS inservice leak and hydrostatic testing..

R.1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Insert: B 3.4-9-01

~~The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).~~ R.1

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. R.1

Insert: B 3.4-9-02

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

(continued)

R.1

INSERT: B 3.4-9-01

LCO 3.4.3, Figure 3.4.3-1, Heatup Limitations for the Reactor Coolant System, Figure, 3.4.3-2, Cooldown Limitations for the Reactor Coolant System, and Figure 3.4.3-3, Hydrostatic and Inservice Leak Testing Limitations for the Reactor Coolant System, contain P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, respectively (Ref. 1).

INSERT: B 3.4-9-02

The happy face icon shown on Figure 3.4.3-1, Figure, 3.4.3-2, and Figure 3.4.3-3, indicates the side of the curve in which operation is permissible. Conversely, the sad face icon indicates the side of the curve in which operation is prohibited.

BASES

BACKGROUND (continued)

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36. PA.1

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

Insert. B 3.4-11-01

The LCO limits apply to all components of the RCS pressure boundary, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure. R.1

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves. PA.1

Insert: B3.4-11-02

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow: A

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

(continued)

PA.1

INSERT: B 3.4-11-01

Figure 3.4.3-1, Heatup Limitations for the Reactor Coolant System, Figure, 3.4.3-2, Cooldown Limitations for the Reactor Coolant System, and Figure 3.4.3-3, Hydrostatic and Inservice Leak Testing Limitations for the Reactor Coolant System, contain P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, respectively. These figures specify the maximum RCS pressure for various heatup and cooldown rates at any given reactor coolant temperature. The figures provide the limiting RCS pressure and reactor coolant temperature combination for reactor coolant temperature heatup rates up to 60°F/hr and reactor coolant temperature cooldown rates up to 100°F/hr. Therefore, heatup rates that exceed 60°F/hr and cooldown rates that exceed 100°F/hr are considered not within the limits of this LCO.

PA.1

INSERT: B 3.4-11-02

Heatup and cooldown limits are specified in hourly increments (i.e., the heatup and cooldown limits are based on the temperature change averaged over a one hour period). Limit lines for cooldown rates between those presented may be obtained by interpolation.

BASES

LCO (continued)

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existence~~s~~, size~~s~~, and orientation~~s~~ of flaws in the vessel material.

PA.1

R.1

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < [500] psig within 36 hours.

Insert: B3.4-14-01

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

(continued)

INSERT: B 3.4-14-01 R.1

Note that LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), will also apply and may require limits for operation that are more restrictive than or supplement this limit.

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time. PA.1

Insert: B3.4-15-01

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-7924-A, April 1975. July 1972
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. ASTM E 185-82, ~~July 1982~~. 70
5. 10 CFR 50, Appendix H.

(continued)

PA.1

INSERT: B 3.4-15-01

Heatup and cooldown limits are specified in hourly increments (i.e., the heatup and cooldown limits are based on the temperature change averaged over a one hour period).

BASES

REFERENCES (continued)	6.	Regulatory Guide 1.99, Revision 2, May 1988.
	7.	ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.5:
"RCS Loops - MODE 3"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops – MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and either:

a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

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

All reactor coolant pumps may not be in operation for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause reduction of the RCS boron concentration; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

RAI 14

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required RCS loop inoperable.	A.1	Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 4.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal.	C.1 Restore required RCS loop to operation. __OR__ C.2 De-energize all control rod drive mechanisms (CRDMs).	1 hour 1 hour
D. Two required RCS loops inoperable. __OR__ No RCS loop in operation.	D.1 De-energize all CRDMs. __AND__ D.2 Suspend all operations involving a reduction of RCS boron concentration. __AND__ D.3 Initiate action to restore one RCS loop to OPERABLE status and in operation.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	12 hours
SR 3.4.5.2	Verify steam generator secondary side actual water levels are ≥ 71% wide range for required RCS loops.	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

NYPA

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required loops are in operation. Verification can be based on flow rate, temperature, or pump status monitoring, which ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the actual secondary side water level is ≥ 71% wide range for each required loop. Depending on plant conditions, either wide range or narrow range SG level instruments may be used to verify this SR is met. Operators may be required to adjust the indicated level to compensate for the effects of SG temperature. If the SG secondary side actual water level is < 71% wide range, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

RAI -17

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

REFERENCES 1. FSAR 14.1.6.

RAI -16

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.5:
"RCS Loops - MODE 3"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops—MODE 3

<CTS>

<3.1.A.1.b>
<DOC M.1>
<DOC L.1>
<DOC M.5>

LCO 3.4.5 [Two] RCS loops shall be OPERABLE, and either:

a. [Two] RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

<3.1.A.1.b.1>
<3.1.A.1.a>
<DOC M.3>

---------------------------NOTE---------------------------
All reactor coolant pumps may ~~be de-energized~~ not be in operation for ≤ 1 hour per 8 hour period provided: (T.1)

a. No operations are permitted that would cause reduction of the RCS boron concentration; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature. R.1

<3.1.A.1.b.1>
<3.1.A.1.b.2>

APPLICABILITY: MODE 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.2>	A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
<DOC M.2>	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<Doc M.2> C. One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal.	C.1 Restore required RCS loop to operation. OR C.2 De-energize all control rod drive mechanisms (CRDMs).	1 hour 1 hour
D. ~~[Two]~~ Required RCS loops inoperable. OR <Doc M.2> No RCS loop in operation.	D.1 De-energize all CRDMs. AND D.2 Suspend all operations involving a reduction of RCS boron concentration. AND D.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately (PA.1) Immediately Immediately (PA.1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<Doc M.4> SR 3.4.5.1 Verify required RCS loops are in operation.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY	
<DOC M.4>	SR 3.4.5.2	Verify steam generator secondary side actual water levels are ≥ [17]% 71% wide range for required RCS loops.	12 hours	DB.2 R.1
<DOC M.4>	SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days	

BASES

APPLICABLE SAFETY ANALYSES (continued)

met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops—MODE 3 satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

The purpose of this LCO is to require that at least [two] RCS loops be OPERABLE. In MODE 3 with the RTBs in the closed position and Rod Control System capable of rod withdrawal, [two] RCS loops must be in operation. [Two] RCS loops are required to be in operation in MODE 3 with RTBs closed and Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an ~~inadvertent control~~ uncontrolled rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents. DB.1

With the RTBs in the open position, or the CRDMs de-energized, the Rod Control System is not capable of rod withdrawal; therefore, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure ~~that safety analyses limits are met~~ redundant decay heat removal capability. PA.1

The Note permits all RCPs to ~~be de-energized~~ not be in operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to ~~perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input~~ Insert: B3.4-22-01 T.1 PA.1

(continued)

PA.1

R.1

INSERT: B 3.4-22-01

permit performance of required tests or maintenance that can only be performed with all reactor coolant pumps not in operation.

BASES

LCO (continued)

values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.

The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the ~~de-energizing~~ stopping of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified ~~is adequate to perform the desired tests,~~ ~~and~~ is acceptable because operating experience has shown that boron stratification is not a problem during this short period with no forced flow. PA.1 R.1 PA.1

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by ~~initial startup~~ test or maintenance procedures: PA.1

a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with RTBs in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level ~~is ≥ [17]% for required RCS loops~~. If the SG secondary side narrow range water level is < [17]%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

Insert: B 3.4-26-01

actual

DB.2

actual

71

wide range

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ~~ensures that safety analyses limits are met. The requirement also~~ ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

REFERENCES

None.

1. FSAR 14.1.6

R.1

INSERT: B 3.4-26-01 R.1

is ≥ 71% wide range for each required loop. Depending on plant conditions, either wide range or narrow range SG level instruments may be used to verify this SR is met. Operators may be required to adjust the indicated level to compensate for the effects of SG temperature.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.6:
"RCS Loops - MODE 4"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops – MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

--------------------------- NOTES ---------------------------

1. All reactor coolant pumps (RCPs) and RHR pumps may not be in operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature. RAI -18

2. No RCP shall be started with any RCS cold leg temperature less than the LTOP arming temperature unless the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)," are met.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable. AND Two RHR loops inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required RHR loop inoperable. AND Two required RCS loops inoperable.	B.1 Be in MODE 5.	24 hours
C. Required RCS or RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1 Suspend all operations involving a reduction of RCS boron concentration. AND C.2 Initiate action to restore one loop to OPERABLE status and in operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side water actual level is ≥ 71% wide range for each required RCS loop.	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

RAI-21

BASES

APPLICABILITY (continued)

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1

If one required RHR loop is OPERABLE and in operation and there are no RCS loops OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the only OPERABLE RHR loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems. NYPA

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status and in operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one RCS or RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the actual secondary side water level is ≥ 71% wide range for each required loop. Depending on plant conditions, either wide range or narrow range SG level instruments may be used to verify this SR is met. Operators may be required to adjust the indicated level to compensate for the effects of SG temperature. If the SG secondary side actual water level is < 71% wide range, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. RA1-21

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.2 (continued)

The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump and associated support systems. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. FSAR Chapter 14.1.6. NYPA

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.6:

"RCS LOOPS - MODE 4"

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
3.1-1	121	121	
3.1-2	179	179	
3.1-3	179	179	
3.1-7	121	121	
3.3-5	53	53	

3. LIMITING CONDITIONS FOR OPERATION

SEE ITS 3.0

For the cases where no exception time is specified for inoperable components, this time is assumed to be zero.

3.1 REACTOR COOLANT SYSTEM

A.2

~~Applicability~~

~~Applies to the operating status of the Reactor Coolant System; operational components; heatup; cooldown; criticality; activity; chemistry and leakage.~~

~~Objective~~

~~To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.~~

Specification

~~A.~~ OPERATIONAL COMPONENTS

~~1.~~ Coolant Pumps

LCO 3.4.6, Note 1.a
Req. Act C.1

a. When a reduction is made in the boron concentration of the reactor coolant, at least one reactor coolant pump or one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation.

SEE ITS 3.4.5

b. (1) When the reactor coolant system T_{avg} is greater than 350°F and electrical power is available to the reactor coolant pumps, and as permitted during special plant evolutions, at least one reactor coolant pump shall be in operation. All reactor coolant pumps may be de-energized for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

(2) When the reactor is subcritical and reactor coolant system T_{avg} is greater than 350°F, control bank withdrawal shall be prohibited unless four reactor coolant pumps are operating.

LCO 3.4.6, Applicability

LCO 3.4.6

LCO 3.4.6, Note 1

~~When the reactor coolant system T_{avg} is greater than 200°F and less than 350°F,~~ Mode 4 ~~and as permitted during special plant evolutions~~ A.4 at least one reactor coolant pump or one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. A.3 All reactor coolant pumps may be de-energized with RHR not in service for up to 1 hour per 8 hour period provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

M.1

Amendment No. ~~48~~, ~~53~~, ~~82~~, ~~84~~, ~~87~~, ~~93~~, ~~98~~, 121

Add SR 3.4.6.1
SR 3.4.6.2
SR 3.4.6.3

M.2

R.1

SEE ITS 3.4.7 ITS 3.4.8

d. When the reactor coolant system T_{avg} is less than 200°F, but not in the refueling operation condition, and as permitted during special plant evolutions, at least one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. This RHR pump may be out of service for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

SEE ITS 3.4.4 ~~ITS 3.4.17~~

e. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

f. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation. R1

g. If the requirements of 3.1.A.1.e and 3.1.A.1.f above cannot be satisfied, the reactor shall be brought to the hot shutdown condition within 1 hour.

SEE ITS 3.4.12

h. A reactor coolant pump (RCP) may not be started (or jogged) when the RCS cold leg temperature (T_{cold}) is at or below 319°F, unless RCS make up is not in excess of RCS losses, and one of the following requirements is met:

(1) The OPS is operable, steam generator pressure is not decreasing, and the temperature of each steam generator is less than or equal to the coldest T_{cold};

Or

(2) The OPS is operable, the temperature of the hottest steam generator exceeds the coldest T_{cold} by no more than 64°F, pressurizer level is at or below 73 percent, and T_{cold} is as per Figure 3.1.A-1;

Or

(3) With OPS inoperable, steam generator pressure is not decreasing, the temperature of each steam generator is less than or equal to the coldest T_{cold}, pressurizer level is at or below 73 percent, and the RCS pressure does not exceed that given by Fig. 3.1.A-2. The pressurizer level must be further restricted per Figures 3.1.A-5 and 3.1.A-6 if operation below 319°F exceeds 8 hours.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.6:
"RCS Loops - MODE 4"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

3.4 REACTOR COOLANT SYSTEM (RCS)

<CTS>

3.4.6 RCS Loops—MODE 4

<3.1.A.1.c>
<3.3.A.6>
<DOC A.3>

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

<3.1.A.1.a>
<DOC M.1>

-----------------------------NOTES-----------------------------

1. All reactor coolant pumps (RCPs) and RHR pumps may ~~be~~ ~~de-energized~~ not be in operation for ≤ 1 hour per 8 hour period provided: T.1

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature. R.1

<3.1.A.1.d>
<DOC A.5>

2. No RCP shall be started with any RCS cold leg temperature ~~≤ [275]°F unless the secondary side water temperature of each steam generator (SG) is ≤ [50]°F above each of the RCS cold leg temperatures.~~ Insert: 3.4-11-01 DB.3

<3.1.A.1.c>
<3.3.A.6>

APPLICABILITY: MODE 4.

ACTIONS

<3.3.A.6.d>
<DOC A.6>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable. AND Two RHR loops inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.3.A.6.d> <DOC L.1> <DOC A.6>	B. One required RHR loop inoperable. AND Two required RCS loops inoperable.	B.1 Be in MODE 5.	24 hours
<3.1.A.1.a> <DOC L.1> <DOC A.6>	C. Required RCS or RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1 Suspend all operations involving a reduction of RCS boron concentration. AND C.2 Initiate action to restore one loop to OPERABLE status and in operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<DOC M.2>	SR 3.4.6.1 Verify one RHR or RCS loop is in operation.	12 hours
<DOC M.2>	SR 3.4.6.2 Verify SG secondary side water actual level~~s~~ ~~are ≥ [17]% for~~ is ≥ 71% wide range for each required RCS loop~~s~~.	12 hours

R.1

DB.2

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.2

actual

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level ~~is ≥ [17]%~~. If the SG secondary side ~~narrow range~~ water level is < [17]%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

DB.2

71

R.1 Insert: B 3.1-31-01

actual

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

and associated support systems

REFERENCES ~~None~~. 1. FSAR Chapter 14.

INSERT: B 3.4-31-01

DB2

R.1

is ≥ 71% wide range for each required loop. Depending on plant conditions, either wide range or narrow range SG level instruments may be used to verify this SR is met. Operators may be required to adjust the indicated level to compensate for the effects of SG temperature.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.7:
"RCS Loops - MODE 5, Loops Filled"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops – MODE 5, Loops Filled

LCO 3.4.7

One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least two steam generators (SGs) shall be ≥ 71% wide range. RAI-24 RAI-22

------------------------- NOTES -------------------------

1. The RHR pump of the loop in operation may not be in operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature. RAI-22

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

3. No reactor coolant pump shall be started with the average of the RCS cold leg temperatures ≤ 319°F unless the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)," are met. NFPA

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR loop inoperable. AND Required SGs secondary side water level not within the limit.	A.1	Initiate action to restore a second RHR loop to OPERABLE status.	Immediately
	OR		
	A.2	Initiate action to restore required SG secondary side water level to within the limit.	Immediately
B. Required RHR loops inoperable. OR No RHR loop in operation.	B.1	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	AND		
	B.2	Initiate action to restore one RHR loop to OPERABLE status and in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	12 hours
SR 3.4.7.2	Verify SG secondary side water level is ≥ 71% wide range in required SGs.	12 hours
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

RAI-22

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops – MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant, via natural circulation (Ref. 1), or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs, via natural circulation (Ref. 1), are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification. The boron concentration in the pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than the rest of the reactor coolant.

Each RHR loop consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves to transfer heat between the RHR heat exchanger and the core. Although either RHR heat exchanger may be credited for either RHR loop, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR loop.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at

(continued)

BASES

BACKGROUND (continued)

least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels ≥ 71% wide range to provide an alternate method for decay heat removal via natural circulation (Ref. 1). RAI-22

When using SGs depending on natural circulation as the backup decay heat removal system in Mode 5, consideration should be given to the potential need for the following: (1) the ability to pressurize and control pressure in the RCS, (2) secondary side water level in the SG relied upon for decay heat removal, (3) availability of a supply of feedwater, and (4) availability of an auxiliary feedwater pump capable of injecting into the relied-upon SGs (Ref.1). RAI-24

During natural circulation, the SGs secondary side water may boil creating the need to release steam through the atmospheric relief valves or other openings that may exist during shutdown conditions. Therefore, consideration should be given to avoiding the potential for pressurization of the SGs secondary side. It is also important to note that during decay heat removal using natural circulation, a MODE change (MODE 5 to MODE 4) could occur due to heat up of the RCS (Ref.1). RAI-24

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops – MODE 5 (Loops Filled) satisfy Criterion 4 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ 71% wide range. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with secondary side water level ≥ 71% wide range. Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation. RAI-22 RAI-22

Note 1 permits all RHR pumps to not be in operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit testing and maintenance that can be performed only when in MODE 5 with no forced circulation. This allowance is acceptable because operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by maintenance or test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction. RAI-22

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during MODE 5 with no forced circulation.

(continued)

BASES

LCO (continued)

Note 3 requires that the reactor coolant pump starting requirements of LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), must be met before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature less than the LTOP arming temperature specified in LCO 3.4.12, Low Temperature Overpressure Protection (LTOP). This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink with forced flow or natural circulation when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

NYPA

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 71% wide range. RAI-22

Loops filled is based on the ability to use the SGs as a backup means of decay heat removal. The RCS loops are considered filled provided that pressurizer level has been maintained ≥10%. The loops are also considered filled following the completion of filling and venting the RCS. The ability to pressurize the RCS to ≥100 psig and to control pressure must be established to take credit for use of the SGs as backup decay heat removal. This is to prevent flashing and void formation at the top of the SG tubes NYPA

(continued)

BASES

APPLICABILITY (continued)

which may degrade or interrupt the natural circulation flow path (Ref. 1). NYPA

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water level < 71% wide range redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. RAI-22 RAI-24 RAI-24

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and in operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring the secondary side water level ≥ 71% wide range ensures an alternate decay heat removal method, via natural circulation, in the event that the second RHR loop is not OPERABLE. Depending on plant conditions, either wide range or narrow range SG level instruments may be used to verify this SR is met. Operators may be required to adjust the indicated level to compensate for the effects of SG temperature. RAI-22

If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level. NYPA

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side water level is ≥ 71% wide range in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience. RAI-22 RAI-24

(continued)

BASES (continued)

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.7:
"RCS Loops - MODE 5, Loops Filled"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

requirements is an administrative change with no adverse impact on safety.

A.4 CTS 3.1.A.1.d specifies that CTS requirements for decay heat removal may be modified "as permitted during special plant evolutions." ITS 3.4.7 deletes this exception to the LCO applicability because it is ambiguous and does not provide any clearly identifiable requirements or allowances. Therefore, deletion of this statement results in no change to the existing requirements. Therefore, this is an administrative change with no impact on safety.

A.5 CTS 3.1.A.1.h establishes requirements for starting reactor coolant pumps (RCPs) when reactor coolant system temperature is below the low temperature overpressure protection (LTOP) enable temperature (i.e., 319°F), ITS LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), includes surveillance requirements that maintain these allowances and requirements (See ITS 3.4.12). ITS LCO 3.4.7, Note 3, is added to ensure that ITS LCO 3.4.12 requirements are met prior to starting RCPs when in Mode 5. The addition of ITS LCO 3.4.7, Note 3, is an administrative change with no adverse impact on safety because it is a cross reference between ITS LCO 3.4.7 and ITS LCO 3.4.12 requirements.

A.6 The combination of CTS 3.1.A.1.d and CTS 3.3.A.7 establish requirements for decay heat removal when the reactor coolant system Tavg is less than 200°F but not in the refueling condition (Mode 5). CTS 3.1.A.1.d and CTS 3.3.A.7 do not make an explicit distinction between Mode 5 with loops filled and Mode 5 with loops not filled; however, with loops not filled a SG is not capable of removing decay heat.

ITS LCO 3.4.7, RCS Loops - Mode 5, Loops Filled, and ITS LCO 3.4.8, RCS Loops - Mode 5, Loops Not Filled, establish requirements consistent with the combination of the two CTS requirements. The primary difference between ITS LCO 3.4.7 and ITS LCO 3.4.8 is that if the RCS loops are filled, then two filled SGs can be credited as an alternate method of decay heat removal in place of an RHR loop that is not operating. This is consistent with a reasonable interpretation of the CTS. Therefore, to satisfy requirements of ITS LCO 3.4.7.

This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification that each RCS loop is operating and/or Operable as required by the LCO. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.3.A.7.b allows an alternate means of decay heat removal to be used in place of one or both RHR loops without any time restrictions as long as the alternate method is capable of maintaining RCS temperature. This is a special allowance that may be used during maintenance, modifications, testing, inspection or repair.

ITS LCO 3.4.7 does not include an allowance for unlimited use of a temporary decay heat removal system as one of the two required decay heat removal systems (although ITS 3.4.7 does permit the use of a SG as the backup decay heat removal system (See ITS 3.4.7, DOC L.1)).

Deletion of CTS 3.3.A.7.b is needed and is acceptable because ITS LCO 3.4.7 provides appropriate allowances for performing required testing and maintenance which could temporarily render one of the two required decay heat removal systems inoperable.

This more restrictive change is acceptable because it does not introduce any operation which is un-analyzed while eliminating the option for unlimited use of a temporary decay heat removal system as one of the two required decay heat removal systems Therefore, this change has no adverse impact on safety.

<u>LESS RESTRICTIVE</u>

L.1 CTS 3.1.A.1.d requires one RHR pump be operating when in Mode 5. CTS 3.3.A.7 requires that two RHR pumps be Operable in Mode 5 but allows the requirements for two Operable RHR pumps in Mode 5 to be suspended during maintenance, modifications, testing, inspection or repair provided that an alternate means of decay heat removal is available and return of the system within sufficient time to prevent exceeding cold shutdown requirements is assured (See ITS 3.4.7, DOC M.3).

ITS LCO 3.4.7 requires one RHR loop be Operable and in operation and either one additional RHR loop or the secondary side water level of at least two steam generators (SG) with the secondary side filled to a level that ensures the tubes are covered. Therefore, ITS 3.4.7 allows two SGs to be used as the redundant decay heat removal capability at any time in Mode 5 when loops are filled. This change is acceptable because of the following: a) the filled SGs may be used as a backup only and ITS 3.4.7 still requires at least one RHR loop operable and one RHR pump in operation; and, b) two filled SGs with filled RCS loops are capable of providing adequate decay heat removal capability in Mode 5 with either forced or natural circulation. Therefore, this change has no adverse impact on safety. Therefore, this change has no adverse impact on safety.

L.2 ITS LCO 3.4.7, Notes 2 and 4, add two new allowances to the requirements for decay heat removal capability in Mode 5.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is Operable and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible. This change is acceptable because the decay heat removal capability function is maintained, the duration of the period without redundant decay heat removal capability is limited to 2 hours, and appropriate required actions are provided if both methods of decay heat removal are lost.

Note 4 allows both RHR loops to be removed from operation during planned heatup to Mode 4 when at least one RCS loop is in operation. This change is acceptable because during a planned heatup to Mode 4 at least one RCS loop is in operation which means that plant status is set for RCS temperature to exceed Mode 5 limits.These changes have no significant adverse impact on safety.

REMOVED DETAIL

LA.1 CTS 3.3.A.7 establishes requirements for decay heat removal capability using RHR pumps in Mode 5 that includes a listing of the principal

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.7:
"RCS Loops - MODE 5, Loops Filled"

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 3.4.7 - RCS Loops – MODE 5, Loops Filled

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 allows two filled SGs and natural circulation in the reactor coolant system to be credited as the backup decay heat removal capability in Mode 5 when the loops are filled. This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because of the following: a) the filled SGs may be used as a backup only and ITS 3.4.7 still requires at least one RHR loop operable and one RHR pump in operation; and, b) two filled SGs with filled RCS loops are capable of providing adequate decay heat removal capability in Mode 5 with either forced or natural circulation. Therefore, this change has no adverse impact on safety.

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

   The proposed 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 of the following: a) the filled SGs may be used as a backup only and ITS 3.4.7 still requires at least one RHR loop operable and one RHR pump in operation; and, b) two filled SGs with filled RCS loops are capable of providing adequate decay heat removal capability in Mode 5 with either forced or natural circulation. Therefore, this change has no adverse impact on safety. Therefore, this change has no adverse impact on safety. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety of the following: a) the filled SGs may be used as a backup only and ITS 3.4.7 still requires at least one RHR loop operable and one RHR pump in operation; and, b) two filled SGs with filled RCS loops are capable of providing adequate decay heat removal capability in Mode 5 with either forced or natural circulation. Therefore, this change has no adverse impact on safety. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed changes do not involve a significant hazards consideration are discussed below.

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

   This change add two new allowances to the requirements for decay heat removal capability in Mode 5. Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is Operable and in operation. Note 4 allows both RHR loops to be removed from operation during planned heatup to Mode 4 when at least one RCS loop is in operation. The Note 2 change will not result in a significant increase in the probability or consequences of an accident previously evaluated because at least one decay heat removal capability is maintained by Note 2, the duration of the period without redundant decay heat removal capability is limited to 2 hours, and appropriate required actions are provided in the LCO if both methods of decay heat removal are lost. The Note 4 change will not result in a significant increase in the probability or consequences of an accident previously

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.7:
"RCS Loops - MODE 5, Loops Filled"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops—MODE 5, Loops Filled

⟨3.1.A.1.d⟩ ⟨3.3.A.7⟩ ⟨DOC A.3⟩ ⟨DOC L.1⟩

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or | R.1

b. The secondary side water level of at least [two] steam generators (SGs) shall be ≥ ~~[17]%~~. 71% wide range

NOTES

⟨3.1.A.1.d⟩ ⟨DOC M.1⟩ ⟨3.1.A.1.a⟩

1. The RHR pump of the loop in operation may be ~~de-energized~~ not be in operation for ≤ 1 hour per 8 hour period provided: T.1

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature. | R.1

⟨DOC L.2⟩

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

⟨DOC A.5⟩ ⟨3.1.A.1.h 3.1.A.1.i 3.1.A.1.j⟩

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ ~~[275]~~ 319°F unless the ~~secondary side water temperature of each SG is ≤ [50]°F above each of the RCS cold leg temperatures~~ Insert 3.4.14-01. R.1

⟨DOC L.2⟩

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

⟨3.1.A.1.d⟩ ⟨3.3.A.7⟩ ⟨DOC A.6⟩

APPLICABILITY: MODE 5 with RCS loops filled.

INSERT: 3.4-14-01

R.1

the requirements of LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), are met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.3.A.7.a> A. One RHR loop inoperable. **AND** <DOC L.1> Required SGs secondary side water level~~s~~ not within the limit~~s~~.	A.1 Initiate action to restore a second RHR loop to OPERABLE status.	Immediately
	OR	
	A.2 Initiate action to restore required SG secondary side water level~~s~~ to within the limit~~s~~.	Immediately (PA.1)
<3.1.A.1.a> <3.3.A.7.a> B. Required RHR loops inoperable. **OR** No RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	AND	
	B.2 Initiate action to restore one RHR loop to OPERABLE status and in operation.	Immediately (PA.1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<DOC M.2> SR 3.4.7.1	Verify one RHR loop is in operation.	12 hours
<DOC M.2> SR 3.4.7.2	Verify SG secondary side water level is ≥ ~~17%~~ 71% wide range in required SGs.	12 hours (DB.2) R.1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

<DOC M.2>

SURVEILLANCE		FREQUENCY
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops—MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) ~~or the component cooling water via the residual heat removal (RHR) heat exchangers~~. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid. (T.2)

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification. Insert: B 3.4-32-01

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels above ~~[17]%~~ 71% wide range to provide an alternate method for decay heat removal via natural circulation (Ref. 1). (T.2)

Insert: B 3.4-32-03

R.1

DB 1

INSERT: B 3.4-32-01

The pressurizer boron concentration is not a concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Each RHR loop consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves to transfer heat between the RHR heat exchanger and the core. Although either RHR heat exchanger may be credited for either RHR loop, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR loop.

INSERT: B 3.4-32-02

(Not Used)

R.

X.1

INSERT: B 3.4-32-03

When using SGs depending on natural circulation as the backup decay heat removal system in Mode 5, consideration should be given to the potential need for the following: (1) the ability to pressurize and control pressure in the RCS, (2) secondary side water level in the SG relied upon for decay heat removal, (3) availability of a supply of feedwater, and (4) availability of an auxiliary feedwater pump capable of injecting into the relied-upon SGs (Ref.1).

During natural circulation, the SGs secondary side water may boil creating the need to release steam through the atmospheric relief valves or other openings that may exist during shutdown conditions. Therefore, consideration should be given to avoiding the potential for pressurization of the SGs secondary side. It is also important to note that during the decay heat removal using natural circulation, a MODE change (MODE 5 to MODE 4) could occur due to heat up of the RCS (Ref.1).

BASES (continued)

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

Insert: B 3.4-33-01

RCS Loops—MODE 5 (Loops Filled) ~~have been identified in the NRC Policy Statement as important contributors to risk reduction~~.

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ [17]%. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with ~~their~~ secondary side water levels ≥ [17]%. Should the operating RHR loop fail, the SGs could be used to remove the decay heat.

71% wide range

via natural circulation

DB.2

not be in operation

T.1

Note 1 permits all RHR pumps to ~~be de-energized~~ ≤ 1 hour per 8 hour period. The purpose of the Note is to permit ~~tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and~~ operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Insert: B 3.4-33-03

PA.1

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by ~~initial startup~~ test procedures:

maintenance or

PA.1

(continued)

INSERT: B 3.4-33-01

PA.1

satisfy Criterion 4 of 10 CFR 50.36.

INSERT: B 3.4-33-02

(Not Used)

INSERT: B 3.4-33-03

PA.1

testing and maintenance that can be performed only when in MODE 5 with no forced circulation. This allowance is acceptable because

BASES

LCO (continued)

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during ~~the only time when such testing is safe and possible~~ MODE 5 with no forced circulation

Note 3 requires that ~~the secondary side water temperature of each SG be ≤ [50]°F above each of the RCS cold leg temperatures~~ Insert: B3.4-34-01 before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ [275]°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. DB3

DB3

Insert: B3.4-34-02

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink with forced flow or natural circulation when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program. X.1

PA.1 | R.1

Insert: B3.4-34-03

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE,

PA.1 | R.1

(continued)

Insert: B3.4-34-04

INSERT: B 3.4-34-01

the reactor coolant pump starting requirements of LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), must be met

INSERT: B 3.4-34-02

less than the LTOP arming temperature specified in LCO 3.4.12, Low Temperature Overpressure Protection (LTOP).

INSERT: B 3.4-34-03

Deleted | R.1

INSERT: B 3.4-34-04

Loops filled is based on the ability to use SGs as a backup means of decay heat removal. The RCS loops are considered filled provided that pressurizer level has been maintained ≥10%. The loops are also considered filled following the completion of filling and venting the RCS. The ability to pressurize the RCS to ≥100 psig and to control pressure must be established to take credit for use of the SGs as backup decay heat removal. This is to prevent flashing and void formation at the top of the SG tubes which may degrade or interrupt the natural circulation flow path (Ref. 1). | R.1

BASES

APPLICABILITY (continued)

or the secondary side water level of at least [two] SGs is required to be ≥ [17]%. R.1

71% wide range

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.~~5~~ 4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.~~6~~ 5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels < [17]%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. R.1

< 71% wide range

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and in operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal. PA.1

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

71% wide range

Verifying that at least two SGs are OPERABLE by ensuring their secondary side ~~narrow range~~ water levels are ≥ [17]% ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

via natural circulation

Insert B3.4-36-01

R.1

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side water level is ≥ [17]% in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

71% wide range

REFERENCES

None.

1. NRC Information Notice 95-35, "Degraded ability of Steam Generators to Remove Decay Heat by Natural Circulation."

T.1

X.1

INSERT: B 3.4-36-01

DC1

Depending on plant conditions, either wide range or narrow range SG level instruments may be used to verify this SR is met. Operators may be required to adjust the indicated level to compensate for the effects of SG temperature.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.8:
"RCS Loops - MODE 5, Loops Not Filled"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops – MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

-----NOTES-----

1. All RHR pumps may not be in operation for ≤ 15 minutes provided:
   a. The core outlet temperature is maintained at least 10°F below saturation temperature.
   b. No operations are permitted that would cause a reduction of the RCS boron concentration; and
   c. No draining operations to further reduce the RCS water volume are permitted.
2. One RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

RAI-25

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops – MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two loops be available to provide redundancy for heat removal.

Each RHR loop consists of one RHR pump and one RHR heat exchanger as well as associated piping and valves to transfer decay heat between the RHR heat exchanger and the core. Although either RHR heat exchanger may be credited for either RHR loop, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR loop. Separate RHR loops may include common piping and valves.

NYPA

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) satisfy Criterion 4 of 10 CFR 50.36.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet redundancy considerations.

Note 1 permits all RHR pumps to not be in operation for ≤ 15 minutes. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short (e.g., station blackout testing) and core outlet temperature is maintained ≥ 10°F below saturation temperature. The Note prohibits boron dilution or draining operations when RHR forced flow is stopped. NYPA

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop when in MODE 5.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

(continued)

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.8:
"RCS Loops - MODE 5, Loops Not Filled"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

3.4 REACTOR COOLANT SYSTEM (RCS)

<3.1.A.1.d> 3.4.8 RCS Loops—MODE 5, Loops Not Filled

<3.3.A.7> LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

{not be in operation} (T.1)

-----NOTES-----

<DOC M.1> 1. All RHR pumps may be de-energized for ≤ 15 minutes ~~when switching from one loop to another~~ provided: (DB.1) R.1

<3.1.A.1.d> a. ~~The~~ core outlet temperature is maintained ≥ 10°F {at least} below saturation temperature. R.1

<3.1.A.1.a> <3.3.A.7> <DOC M.1> b. No operations are permitted that would cause a reduction of the RCS boron concentration; and

c. No draining operations to further reduce the RCS water volume are permitted.

<DOC L.1> 2. One RHR loop may be inoperable for ≤ {up to} 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

<3.1.A.1.d> <3.3.A.7> <DOC A.5> APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.3.A.7.a> A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

BASES

{not be in operation}

T.1

DB.1

LCO (continued)

Note 1 permits all RHR pumps to ~~be de-energized~~ for ≤ 15 minutes ~~when switching from one loop to another~~. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short [and core outlet temperature is maintained ≥ 10°F below saturation temperature]. The Note prohibits boron dilution or draining operations when RHR forced flow is stopped. (e.g., Station blackout testing) DB.1

≥

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop ~~during the only time when these tests are safe and possible~~. {when in Mode 5}

PA.1

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and (4)
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6). (5)

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

{loops}

(continued)

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.9:
"Pressurizer"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 58.3%; and RAI 26

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW and capable of being powered from an emergency power supply. RAI 27

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours 12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 58.3%.	12 hours
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is ≥ 150 kW.	24 months

RAI 26

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling

(continued)

BACKGROUND (continued)

margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

Pressurizer heaters are powered from either the offsite source or the diesel generators (DGs) through the four 480V vital buses as follows: bus 2A (DG 31) supports 485 kW of pressurizer heaters; bus 3A (DG 31) supports 555 kW of pressurizer heaters; bus 5A (DG 33) supports 485 kW of pressurizer heaters; and, bus 6A (DG 32) supports 277 kW of pressurizer heaters.

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. The required pressurizer level of ≤ 58.3% is the analytical limit used as an initial condition in the accident analysis. An additional margin should be allowed for instrument error. RAI 26

Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

(continued)

BASES (continued)

LCO

The LCO requirement for the pressurizer to be OPERABLE with water level less than or equal to 58.3%, ensures that a steam bubble exists. The required pressurizer level of ≤ 58.3% is the analytical limit used as an initial condition in the accident analysis. An additional margin of approximately 7% should be allowed for instrument error (i.e., the indicated level should not exceed 51.3%). RAI 26

Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW, capable of being powered from either the offsite power source or the emergency power supply. Each of the 2 groups of pressurizer heaters should be powered from a different DG to ensure that the minimum required capacity of 150 kW can be energized during a loss of offsite power condition assuming the failure of a single DG. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The value of 150 kW is sufficient to maintain pressure and is dependent on the heat losses. RAI -27

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from an

(continued)

BASES

APPLICABILITY (continued)

emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions.

NYPA

If the pressurizer water level is not within the limit, action must be taken to place the plant in a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering that the redundant heater group is still available and the low probability of an event during this period. Pressure control may be maintained during this time using remaining heaters.

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done separately by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of 24 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

REFERENCES

1. FSAR, Section 14.

2. NUREG-0737, November 1980.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.9:
"PRESSURIZER"

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
3.1-4	170	170	
3.1-8	179	179	
3.1-25	149	149	

SEE ITS 3.1.3 ITS 3.1.8 ITS 3.4.2

C. MINIMUM CONDITIONS FOR CRITICALITY

1. Except during low power physics test, the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive.

2. This section intentionally deleted.

3. At all times during critical operation, the lowest loop T_{avg} shall be no lower than 540 °F.
   a. If T_{avg} is less than 540°F when the reactor is critical, restore T_{avg} to $\geq$ 540 °F within 15 minutes or be in hot shutdown within the following 15 minutes.

LCO 3.4.9 Applicability

4. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ until normal water level is established in the pressurizer. Mode 1, 2, 3 M.2 ≤ 58.3% L.1 R.1

Basis

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.[1][2] The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.[1][2] Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions. A.1

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of an increase in moderator temperature. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical except when T_{avg} is $\geq$ 540 °F provides assurance that an overpressure event will not occur whenever the reactor vessel is in the nil-ductility temperature range and that the reactor is operated within the bounds of the safety analyses. The safety analyses, which assume a critical temperature of 547 °F, are applicable for critical temperatures as low as 540 °F. Heatup to this temperature will be accomplished by operating the reactor coolant pumps. The Surveillance requirement to support this specification is provided in Table 4.1-1 item no. 4.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the reactor coolant not be solid when criticality is achieved.

References

1. FSAR Table 3.2-1

2. FSAR Figure 3.2-9

Add Condition A and associated Req. Acts M.3

Amendment No. ~~34~~, ~~109~~, ~~121~~, 149

Add SR 3.4.9.1 M.4

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.9: "Pressurizer"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.4.9 - Pressurizer

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.

MORE RESTRICTIVE

M.1 CTS 3.1.A.3 requires at least 150 kW of pressurizer heaters that are capable of being energized during a loss of offsite power condition so that natural circulation can always be maintained during hot shutdown. CTS 3.1.A.3.a provides an allowable out of service time of 72 hours if the required heater capacity is not Operable.

ITS LCO 3.4.9 requires 2 groups of pressurizer heaters and that each of these groups must have a capacity of 150 kW and capable of being powered from an emergency power supply. The LCO Bases specify that the intent is that each required group must be powered from a different safeguards power train (i.e., diesel generator (DG). In conjunction with this change, LCO 3.4.9. Required Action B.1, provides an allowable out of service time of 72 hours if one of the two required heater groups is not Operable. Furthermore, although not stated as an Action for ITS LCO 3.4.9, entry into LCO 3.0.3 is required if neither group of pressurizer heaters is Operable.

This change is needed because 150 kW of pressurizer heater capacity must be available in Modes 1, 2 and 3 (See ITS 3.4.9, DOC M.2) to support decay heat removal using natural circulation following a loss of offsite power. Requiring 2 groups of pressurizer heaters and that each group is powered from a separate DG ensures that the single failure of a DG will not result in a loss of the required pressurizer heater capacity.

This more restrictive change is acceptable because it does not introduce any operation that is un-analyzed while ensuring the required pressurizer heater capacity will be available following a loss of offsite power with concurrent failure of one DG. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.1.A.3 specifies that the pressurizer must be Operable with the specified heater capacity whenever the reactor is above the hot shutdown condition (Modes 1 and 2). CTS 3.1.C.4 requires that the pressurizer normal water level must be maintained (See ITS 3.4.9, DOC L.1) whenever the reactor is not subcritical by at least 1% Δk (Modes 1 and 2).

ITS LCO 3.4.9, Applicability, requires the pressurizer Operable with the level below the specified maximum and with required heater capacity whenever the plant is in Modes 1, 2 and 3. In conjunction with this change, ITS 3.4.9, Required Actions A.2 and C.2, are added to require that the plant be placed outside Applicability (i.e., the plant must be placed in Mode 4) if requirements are not met.

This change, requiring pressurizer Operability in Mode 3, is needed because pressurizer Operability in Mode 3 will prevent solid water operation during heatup and cooldown and during other operational perturbations (e.g., RCP starts) that could cause rapid pressure increases if the pressurizer is solid.

This change is acceptable because it does not introduce any operation that is un-analyzed while requiring that the pressurizer be available for pressure control during heatup and cooldown and during other operational perturbations (e.g., RCP starts) that could cause rapid pressure increases if the pressurizer is solid. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.1.C.4 requires that the pressurizer normal water level must be maintained (See ITS 3.4.9, DOC L.1) whenever the reactor is not subcritical by at least 1% Δk (Modes 1 and 2); however, no Actions are specified if this requirement is not met (although if pressurizer water level reached the ITS LCO 3.4.9 limit, a reactor trip on Pressurizer Water Level-High would occur).

ITS LCO 3.4.9, Required Actions A.1 and A.2, are added to require that a reactor must be placed in Mode 4 within 12 hours if pressurizer water level cannot be maintained within the specified limit.

This change is needed to supplement the reactor trip on Pressurizer Water Level-High and require that the plant be placed outside the LCO Applicability (i.e., the plant must be placed in Mode 4) in addition to the reactor shutdown caused by the reactor trip on Pressurizer Water Level-High to prevent solid water operation during heatup and cooldown and during other operational perturbations (e.g., RCP starts) that could cause rapid pressure increases if the pressurizer is solid in Mode 3. This change is acceptable because it does not introduce any operation that is un-analyzed. Therefore, this change has no adverse impact on safety.

M.4 CTS 3.1.A.3 requires a specified pressurizer heater capacity must be available whenever the reactor is above the hot shutdown condition (See ITS 3.4.9, DOC M.2). CTS 3.1.C.4 requires that a specified pressurizer water level must be maintained (See ITS 3.4.9, DOC L.1) whenever the reactor is not subcritical by at least 1% Δk (See ITS 3.4.9, DOC M.2). However, no surveillance requirements are established to periodically verify these requirements are met.

ITS SR 3.4.9.1 is added to verify every 12 hours that pressurizer level is within the required limit. The Frequency of 12 hours is considered adequate because the limit is enforced by the reactor trip on Pressurizer Water Level-High.

ITS SR 3.4.9.2 is added to demonstrated every 24 months that the specified pressurizer heater capacity is available by checking the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of 24 months is considered adequate to detect heater degradation because the have exhibited a high degree of reliability and these heaters are used during normal operation.

These changes are needed to require periodic verification that the requirements of ITS LCO 3.4.9 are met.

These changes are acceptable because they do not introduce any operation that is un-analyzed while requiring periodic verification that pressurizer operation is within specified limits. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.1.C.4 requires normal water level be established in the pressurizer prior to reactor criticality (See ITS 3.4.9, DOC M.2).

ITS LCO 3.4.9 requires that pressurizer water level be less than or equal to 58.3% in Mode 1, 2 and 3 (See ITS 3.4.9, DOC M.2).

Replacing the requirement to maintain pressurizer level in the normal range with a requirement to maintain pressurizer level less than or equal to 58.3% is needed and is acceptable because a pressurizer level of ≤ 58.3% is the analytical limit used as an initial condition in the accident analysis. The Bases include the clarification that an additional margin of approximately 7% should be allowed for instrument error (i.e., the indicated level should not exceed 51.3%). Additionally, the upper limit on pressurizer level ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.9:
"Pressurizer"

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 3.4.9 - Pressurizer

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 replaces the requirement to maintain pressurizer level in the normal range when critical to a requirement to maintain pressurizer water level less than or equal to 58.3%.

   This change will not result in a significant increase in the probability or consequences of an accident previously evaluated because a pressurizer level of ≤ 58.3% is the analytical limit used as an initial condition in the accident analysis. The Bases include the clarification that an additional margin of approximately 7% should be allowed for instrument error (i.e., the indicated level should not exceed 51.3%). Additionally, the upper limit on pressurizer level ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Additionally, the upper limit on pressurizer level ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

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 pressurizer level will be maintained in the normal operating range. This change is consistent with the high pressurizer water level reactor trip that protects the pressurizer safety valves against water relief. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety because a pressurizer level of ≤ 58.3% is the analytical limit used as an initial condition in the accident analysis. The Bases include the clarification that an additional margin of approximately 7% should be allowed for instrument error (i.e., the indicated level should not exceed 51.3%). Additionally, the upper limit on pressurizer level ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Additionally, the upper limit on pressurizer level ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.9:
"Pressurizer"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

<3.1.A.3> LCO 3.4.9 The pressurizer shall be OPERABLE with:

58.3 |R.1

<3.1.C.4><DOC L.1> a. Pressurizer water level ≤ [92]%; and 150

|R.1

<3.1.A.3><DOC M.1> b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ [125] kW [and capable of being powered from an emergency power supply].

DB1

<3.1.A.3>
<3.1.C.4>
<DOC M.2>

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.3>	A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open.	6 hours
		<u>AND</u>	
<DOC M.2>		A.2 Be in MODE 4.	12 hours
<3.1.A.3.a> <DOC M.1>	B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
<3.1.A.3.a>	C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
		<u>AND</u>	
<DOC M.2>		C.2 Be in MODE 4.	12 hours

| R.1

58.3

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
<DOC M.4>	SR 3.4.9.1 Verify pressurizer water level is ≤ ~~[92]~~ 58.3%.	12 hours	
<DOC M.4>	SR 3.4.9.2 Verify capacity of each required group of pressurizer heaters is ≥ ~~[125]~~ 150 kW.	~~92 days~~ 24 months	T.1
	~~SR 3.4.9.3 Verify required pressurizer heaters are capable of being powered from an emergency power supply.~~	~~[18] months~~	

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, ~~and their controls~~ and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to

(continued)

BASES Insert: B 3.4-41-01

BACKGROUND (continued)

a loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

Insert: B 3.4-41-04

Insert: B3.4-41-02

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. R.1

Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

10 CFR 50.36

The maximum pressurizer water level limit satisfies Criterion 2 of ~~the NRC Policy Statement~~. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO. T.2

LCO

58.3%

water level less than or equal to

Insert B3.4-41-04

The LCO requirement for the pressurizer to be OPERABLE with ~~a water volume < [1240] cubic feet, which is equivalent to [92]%, ensures that a steam bubble exists.~~ Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. DB.1 R.1

150

Insert: B3.4-41-03

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ [125] kW, capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.9 - Pressurizer

INSERT: B 3.4-41-01

DA.1

Pressurizer heaters are powered from either the offsite source or the diesel generators (DGs) through the four 480 V vital buses as follows: bus 2A (DG 31) supports 485 kW of pressurizer heaters; bus 3A (DG 31) supports 555 kW of pressurizer heaters; bus 5A (DG 33) supports 485 kW of pressurizer heaters; and, bus 6A (DG 32) supports 277 kW of pressurizer heaters.

INSERT: B 3.4-41-02

DB.1

, which ensures that a steam bubble exists in the pressurizer,

INSERT: B 3.4-41-03

Each of the 2 groups of pressurizer heaters should be powered from a different DG to ensure that the minimum required capacity of 150 kW can be energized during a loss of offsite power condition assuming the failure of a single DG. R.1

INSERT: B 3.4-41-04

LE.1

The required pressurizer level of ≤ 58.3% is the analytical limit used as an initial condition in the accident analysis. An additional margin of approximately 7% should be allowed for instrument error (i.e., the indicated level should not exceed 51.3%).

R.1

BASES

LCO (continued)

(150)

conditions, a wide margin to subcooling can be obtained in the loops. The ~~exact design~~ value of 125 kW ~~is derived~~ from the ~~use of seven heaters rated at 17.9 kW each. The amount needed~~ to maintain pressure is dependent on the heat losses.

is sufficient

and

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters, capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. ~~Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level—High Trip.~~ 1R.1

If the pressurizer water level is not within the limit, action must be taken to ~~restore the plant to operation within the bounds of the safety analyses~~. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the unit out of the applicable MODES. T.2

Insert: B3.4-42-01

(continued)

INSERT: B 3.4-42-01

place the plant in a MODE in which the LCO does not apply.

BASES

ACTIONS

A.1 and A.2 (continued)

~~and restores the unit to operation within the bounds of the safety analyses.~~ T.2

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

Insert: B3.4-43-01

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering ~~the anticipation that a demand caused by loss of offsite power would be unlikely in this period.~~ Pressure control may be maintained during this time using ~~normal station powered~~ remaining heaters.

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, 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 MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours ~~corresponds to verifying the parameter each shift. The 12 hour interval~~ has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within

(continued)

INSERT: B 3.4-43-01

that the redundant heater group is still available and the low probability of an event during this period.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 (continued)

Insert: B3.4-44-01

T.2

safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

separately

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of ~~92 days~~ 24 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

~~SR 3.4.9.3~~

~~This SR is not applicable if the heaters are permanently powered by Class 1E power supplies.~~

~~This Surveillance demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.~~

REFERENCES

1. FSAR, Section ~~[ ]~~ 14.
2. NUREG-0737, November 1980.

INSERT: B 3.4-44-01

of ensuring that a steam bubble exists in the pressurizer

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.9: "Pressurizer"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.4.9 - Pressurizer

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. 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.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-93 (WOG-19), which changes the frequency of pressurizer heater testing (SR 3.4.9.2)from 92 days to 24 months. This change is acceptable because the heaters are normally in operation and significant degradation will be detected. This change is in accordance with Section 6.6 of NUREG-1366.

T.2 This change incorporates Generic Change TSTF-162 (WOG-68), which explains the bases for the maximum pressurizer water level limit. This change is needed to properly explain that the maximum pressurizer water level limit is based on ensuring that a steam bubble exists in the pressurizer. The maximum pressurizer water level is not explicitly credited in any safety analysis.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.10:
"Pressurizer Safety Valves"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings set ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 319°F. RAI 30 RAI 31

-----NOTE-----
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperature ≤ 319°F.	6 hours 12 hours

RAI-31 RAI-30

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be $\geq$ 2460 psig and $\leq$ 2510 psig.	In accordance with the Inservice Testing Program

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine without a direct reactor trip or any other control. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves; or an increase in the pressurizer relief tank temperature or level; or actuation of acoustic monitors.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ 319°F (i.e., less than the LTOP arming temperature specified in LCO 3.4.12) and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." RAI-30 RAI-31

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

(continued)

BASES

APPLICABILITY (continued)

MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when any RCS cold leg temperature is ≤ 319°F (i.e., when LCO 3.4.12 is applicable) or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head removed. RAI-30 RAI-31

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from industry experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, 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 and to MODE 4 with any RCS cold leg temperature ≤ 319°F (i.e., where LCO 3.4.12 is applicable) within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner RAI-30 RAI-31

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

and without challenging plant systems. With any of the RCS cold leg temperatures ≤ 319°F (i.e., when LCO 3.4.12 is applicable) overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves. RAI-30 RAI-31

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. FSAR, Chapter 14.
3. WCAP-7769, Rev. 1, June 1972.
4. ASME, Boiler and Pressure Vessel Code, Section XI.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.10:
"Pressurizer Safety Valves"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

nominal 2485 psig setpoint during the Surveillance to allow for drift during the SR interval. This is needed and is acceptable because the pressurizer safety valves satisfy safety analysis assumptions and meet ASME Code requirements if the setpoint is determined to be ± 3% at the end of the surveillance interval. Therefore, the pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves must be reset to ± 1% during the Surveillance to allow for drift during the SR interval.

This is an administrative change with no impact on safety because this practice (i.e., pressurizer safety valve setpoint is ± 3% for Operability but must be reset to ± 1% during the SR to allow for drift) is consistent with the overpressure analysis, current IP3 practice and the requirements of the ASME, Boiler and Pressure Vessel Code, Section XI.

A.4 CTS Table 4.1-3, Note to Pressurizer Safety Valve Frequency, specifies that the safety valve setpoint test due May 1996 may be deferred until the next refueling outage but no later than May 31, 1997. This note is deleted because the allowance provided has expired. This is an administrative change with no impact on safety.

A.5 CTS 3.1.A.2.b specifies that "all" pressurizer code safety valves must be Operable with a corresponding statement in the CTS Bases regarding the capacity of the three pressurizer code safety valves. ITS 3.4.10 requires that three pressurizer code safety valves must be Operable. This is an administrative change with no impact on safety because the IP3 design includes only three pressurizer code safety valves.

MORE RESTRICTIVE

M.1 CTS 3.1.A.2.b specifies that pressurizer code safety valves must be Operable above the cold shutdown condition except during reactor coolant system hydrostatic tests. ITS LCO 3.4.10 maintains the requirement that pressurizer code safety valves must be Operable during normal plant operation (See ITS LCO 3.4.10, DOC L.1) but exception during reactor coolant system hydrostatic tests is deleted. This change is acceptable because current Section XI of the ASME Boiler and Pressure Vessel Code

DISCUSSION OF CHANGES
ITS SECTION 3.4.10 - Pressurizer Safety Valves

LESS RESTRICTIVE

L.1 CTS 3.1.A.2.b specifies that pressurizer code safety valves must be Operable above the cold shutdown condition (i.e., Modes 1, 2, 3 and 4).

ITS LCO 3.4.10 specifies that pressurizer code safety valves must be Operable in Modes 1, 2, and 3, and in Mode 4 but with all RCS cold leg temperatures > 319°F (i.e., above the Low Temperature Overpressure Protection (LTOP) arming temperature). Therefore, ITS LCO 3.4.10 eliminates the requirement for pressurizer code safety valve OPERABILITY when ITS LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), governs overpressure protection requirements for the reactor coolant system.

This change is acceptable because RCS overpressure protection required by ITS LCO 3.4.12, Low Temperature Overpressure Protection, will ensure adequate protection of the RCS pressure boundary without the use of pressurizer safety valves whenever the RCS is below the LTOP arming temperature. This change has no impact on safety because ITS LCO 3.4.10 and 3.4.12 ensure that RCS overpressure protection consistent with safety analysis assumptions is provided at all times.

L.2 CTS 3.1.A.2 establishes requirements for the OPERABILITY of pressurizer code safety valves but does not specify any required action if this LCO is not met.

ITS LCO 3.4.10, Conditions A and B, establishes required actions when one or more pressurizer safety valves are not operable. Specifically, Condition A requires that with one pressurizer safety valve inoperable, restoration must take place within 15 minutes. This change is needed because an inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary. Condition B requires that if two or more pressurizer safety valves are inoperable or if the requirements of Required Action A.1 cannot be met, then the plant must be brought to a Mode in which the requirement does not apply (i.e., below the LTOP protection arming temperature). This change is needed because if there is less than the required overpressure protection (setpoint or capacity), then the RCS can be protected only by reducing the RCS energy (core power and pressure) which lowers the 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 3.4.10 and ITS 3.4.12 maintain the requirements for RCS overpressure protection. Therefore, requirements for pressurizer code safety valves when below the LTOP arming temperature can be maintained in the FSAR with no significant adverse impact on safety.

LA.2 CTS Table 4.1-3, Item 3, Pressurizer Safety Valves, requires verification of the setpoints every 24 months.

ITS SR 3.4.10.1 maintains the requirement to verify the Operability of pressurizer safety valves including setpoint verification; however, the Frequency is specified as in accordance with the Inservice Test (IST) Program. The IST program requires that pressurizer safety valves are tested every 5 years. This requirement is different from the current frequency of 24 months, but is in accordance with the IP3 approved IST program.

This change is needed and is acceptable because the IST program is required by ITS 5.5.7 and provides controls for inservice testing of all ASME Code Class 1, 2, and 3 components. Specifically, ITS 5.5.7, Inservice Testing Program (IST), requires establishing and maintaining a program for inservice testing of ASME Code Class 1, 2, and 3 components at frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code. Additionally, 10 CFR 50.55a(f) already provides the regulatory requirements for this IST Program, and specifies that ASME Code Class 1, 2, and 3 pumps and valves are covered by an IST Program.

ITS LCO 3.4.10 will still require that pressurizer safety valves must be operable and set within specific limits and ITS SR 3.4.10.1 will still require periodic verification of Operability. These requirements, in conjunction with the IST Program required by ITS 5.5.7, provide a high degree of assurance that safety valves will be tested and maintained to ensure pressurizer safety valve Operability. Additionally, ITS 5.5.7, Inservice Testing Program (IST), requirements and 10 CFR 50.55a(f) ensure adequate change control and regulatory oversight for any changes to the existing requirements. Therefore, requirements to test pressurizer safety valves can be maintained in the IST program with no significant adverse impact on safety.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.10:
"Pressurizer Safety Valves"

PART 5:

NUREG-1431
Annotated to show differences between NUREG-1431 and ITS

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

<3.1.A.2.b>
<3.1.A.2.c>

LCO 3.4.10 [Three] pressurizer safety valves shall be OPERABLE with lift settings ≥ [2460] psig and ≤ [2510] psig.

set

PA.1

<3.1.A.2.b>
<DOC L.1>
<DOC M.1>

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > ~~[275]~~ 319 °F.

R.1

<3.1.A.2.b>
<DOC M.2>

NOTE

The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for [54] hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
<DOC L.2>	A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes	
<DOC L.2>	B. Required Action and associated Completion Time not met. **OR** Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. **AND** B.2 Be in MODE 4 with any RCS cold leg temperatures ≤ ~~[275]~~ 319 °F.	6 hours 12 hours	R.1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed ~~pop type~~, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), [2735] psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, ~~380,000~~ 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine without a direct reactor trip or any other control. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves ~~or~~ ; or, increase in the pressurizer relief tank temperature or level ; or, actuation of acoustic monitors.

DB.1

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ~~≤ [275]°F~~ and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

Insert B 3.4-45-01

R.1

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

Insert: B3.4-45-02

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

(continued)

INSERT: B 3.4-45-01 R.1

≤ 319°F (i.e., less than the LTOP arming temperature specified in LCO 3.4.12)

INSERT: B 3.4-45-02

Although the pressurizer safety valves must be set to ± 1% during the Surveillance, the pressurizer safety valves satisfy safety analysis assumptions and meet ASME Code requirements if the setpoint is determined to be ± 3% at the end of the surveillance interval. Therefore, the pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves must be reset to ± 1% during the Surveillance to allow for drift.

BASES

BACKGROUND (continued)

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

Insert: B3.4-46-03

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of [three] safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Loss of normal feedwater;

e. Loss of all AC power to station auxiliaries; and

f. Locked rotor.

Detailed analyses of the above transients may be are contained in Reference 2. Safety valve actuation ~~is~~ required in events a, b, c, e and f ~~c, d, and e~~ (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions. DB.1

Insert: B3.4-46-01

Pressurizer safety valves satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

The [three] pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig.

Insert: B3.4-46-02 PA.1

(continued)

BASES

LCO (continued)

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of [three] valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when all any RCS cold leg temperatures ~~are~~ is ≤ [275]°F Insert B3.4-47-01 or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head ~~detensioned~~ removed. | R.1

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The [54] hour exception is based on 18 hour outage time for each of the [three] valves. The 18 hour period is derived from ~~operating~~ industry experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection ~~System~~. An inoperable safety valve

(continued)

INSERT: B 3.4-47-01

≤ 319°F (i.e., when LCO 3.4.12 is applicable)

BASES

ACTIONS

A.1 (continued)

coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, 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 and to MODE 4 with any RCS cold leg temperatures ≤ [275]°F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below [275]°F, overpressure protection is provided by ~~the~~ LTOP ~~System~~. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by [three] pressurizer safety valves. R.1

Insert B3.4-48-01

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. FSAR, Chapter ~~[15]~~. 14

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.10 - Pressurizer Safety Valves

INSERT: B 3.4-48-01

≤ 319°F (i.e., when LCO 3.4.12 is applicable)

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.10:
"Pressurizer Safety Valves"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.4.10 - Pressurizer Safety Valves

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. 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 (Not used)

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.11:
"Pressurizer Power Operated Relief Valves (PORVs)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are nitrogen operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal and alternate pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

NYPA

Electrical power needed to support the PORVs, their block valves, and their controls is supplied from the vital buses that normally receive power from offsite power sources, but is also capable of being supplied from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

(continued)

BASES

BACKGROUND (continued)

The plant has two PORVs, each having a design relief capacity of 179,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint following a step reduction of 50% of full load with steam dump and automatic reactor control operation. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal and alternate pressurizer spray are not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs or auxiliary spray are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical (Ref. 2). By assuming PORV manual actuation, the DNBR calculation is more conservative although not required to meet safety limits. As such, this actuation is not required to mitigate these events, and PORV automatic operation is not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

(continued)

BASES

LCO (continued)

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage. Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4, 5 and 6 with the reactor vessel head in place when both pressure and core energy are decreased and the pressure surges become much less significant. LCO 3.4.12 addresses the PORV requirements in these MODES. RAI-40

(continued)

BASES (continued)

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3. This exception to LCO requirements is normally used to perform cycling of the PORVs or block valves to verify their OPERABLE status because testing is not performed in lower MODES.

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., excessive seat leakage). In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valve is required to be closed, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. RAI 40

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

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

If one PORV is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Time of 1 hour is reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 7 days is provided to restore the inoperable PORV to OPERABLE status. If the PORV

(continued)

BASES

ACTIONS

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

cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in the closed position (i.e., switch in manual control). The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 7 days to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an overpressure event if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 7 days, the power will be restored to the PORV. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D. RAI 40

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then 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 and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

E.1, E.2, E.3 and E.4

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If no PORVs are restored within the Completion Time, then 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 and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

F.1 and F.2

If more than one block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control (i.e., closed position) and restore at least one block valve within 2 hours. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply.

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12. | RAI 40

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is important because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an inoperable PORV that is not capable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 7 days, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status.

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice.

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. FSAR, Section 14.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.4.11:
"Pressurizer Power Operated Relief Valves (PORVs)"

PART 5:

NUREG-1431
Annotated to show differences between
NUREG-1431 and ITS

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

DB.1 | R.1

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

nitrogen

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

and alternate

DB.1 | R.1

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

Electrical power needed to support the

The PORVs, their block valves, and their controls ~~are~~ ~~powered~~ from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

is supplied

DB.1

is

supplied

R.1

design

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure—High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition,

179,000

and automatic reactor control operation

(continued)

BASES

R.1

BACKGROUND (continued)

the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." DB.1

and alternate

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator. (Ref. 2)

are

or auxiliary spray

modeled

DB.1 | R.1

The PORVs are ~~used~~ in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV manual actuation, ~~the primary pressure remains below the high pressurizer pressure trip setpoint; thus~~, the DNBR calculation is more conservative. ~~Events that assume this condition include a turbine trip and the loss of normal feedwater (Ref. 2)~~. T.1

although not required to meet safety limits

Insert: B 3.4-51-01

Pressurizer PORVs satisfy Criterion 3 of ~~the NRC Policy Statement~~.

10 CFR 50.36

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. ~~The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage.~~ Satisfying the LCO helps minimize challenges to fission product barriers. T.1

Insert: B 3.4-51-02

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.4.11 - Pressurizer Power Operated Relief Valves (PORVs)

INSERT: B 3.4-51-01

As such, this actuation is not required to mitigate these events, and PORV automatic operation is not an assumed safety function.

INSERT: B 3.4-51-02

An OPERABLE block valve may be either open, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 ~~to minimize challenges to the pressurizer safety valves~~. T.1

Insert: B 3.4-52-01

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in ~~MODE 4~~ when both pressure and core energy are decreased ~~and the pressure surges become much less significant~~. ~~The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place.~~ LCO 3.4.12 addresses the PORV requirements in these MODES. T.1 R.1

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES. because

• This exception to LCO requirements is normally used

A.1

may be associated is required to T.1

Insert. B 3.4-52-02

~~With the~~ PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves ~~should~~ be closed, but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. ~~Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a~~ T.1

(continued)

INSERT: B 3.4-52-01

for manual actuation to mitigate a steam generator tube rupture event.

INSERT: B 3.4-52-02

(e.g., excessive seat leakage). In this condition,

BASES

ACTIONS

A.1 (continued)

~~small break LOCA. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve.~~ This ~~Condition is only intended to~~ permit operation of the plant ~~for a limited period of time not to exceed~~ the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. ~~Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering startup (MODE 2).~~ 2 until T.1

R.1

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

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

If one ~~[or two]~~ PORV~~[s]~~ is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Time~~s~~ of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional ~~72 hours~~ is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D. 3

is

7 days

X.1 R.1

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in ~~manual control~~. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE

the closed position (i.e., switch in manual control)

R.1 PA.1

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs are not capable of mitigating an overpressure event ~~when placed in manual control~~. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the ~~PORV restored, to OPERABLE status~~. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

R.1 X.1 7days

may not be

if the inoperable block valve is not full open

R.1 X.1 7days

T.1

Ts

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then 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 and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

automatic

R.1 T.1

E.1, E.2, E.3, and E.4

If more than one PORV is inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time

(continued)

BASES

ACTIONS

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

to correct the situation. ~~If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two [or three] PORVs inoperable.~~ If no PORVs are restored within the Completion Time, then 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 and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12.

T.1 R.1

F.1, F.2, and ~~F.3~~

If more than one block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control (i.e, closed position) and restore at least one block valve within 2 hours ~~[and restore the remaining block valve within 72 hours]~~. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

PA.1 R.1

G.1 and G.2

If the Required Actions of Condition F are not met, then 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 and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, ~~maintaining~~ automatic PORV OPERABILITY may be required. See LCO 3.4.12.

T.1 R.1

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is ~~of~~ important, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an ~~otherwise~~ inoperable PORV that is not capable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is ~~72 hours~~ 7 days, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status ~~(i.e., completion of the Required Actions fulfills the SR)~~.

T.1 R.1 X.1 R.1

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of ~~[18]~~ 24 months is based on a typical refueling cycle and industry accepted practice.

~~SR 3.4.11.3~~

~~Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.~~

~~SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.4.11.4 (continued)~~

~~The Surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. FSAR, Section ~~[15.2]~~. 14
3. ASME, Boiler and Pressure Vessel Code, Section XI.