Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination	RAI 4 & 5
	AND			
	A.2	Determine QPTR after achieving equilibrium conditions from a THERMAL POWER reduction per Required Action A.1.	Once per 12 hours	
	AND			
	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1. **AND** Once per 7 days thereafter	RAI 5
			(continued)	

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
	<u>AND</u>		
A. (continued)	A.4	Re-evaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		
	A.5	--------NOTES-------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed whenever Required Action A.5 is performed. -------------------- Normalize excore detectors to restore QPTR to within limits.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		(continued)

RAI 5

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.6	--------NOTE-------- Perform Required Action A.6 only after Required Action A.5 is completed. -------------------- Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

RV 5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	-----------------NOTES-------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. -- Verify QPTR is within limit by calculation.	7 days
SR 3.2.4.2	-----------------NOTE-------------------- Not required to be performed until 24 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. -- Verify QPTR is within limit using the movable incore detectors.	24 hours

RAI-5

RAI-5

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 2);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 225 calories/gram for non-irradiated fuel and 200 calories/gram for irradiated fuel (Ref. 3); and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 4).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36.

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

(continued)

BASES (continued)

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of the QPTR determination, if necessary, to comply with the decreases in maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit. RAI 5

A.2

After completion of Required Action A.1, the QPTR may still exceed the specified limit. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow. RAI 5

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. RAI 5 Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. RAI 5

(continued)

BASES

ACTIONS

A.3 (continued)

If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR. RAI 5

(continued)

BASES

ACTIONS

A.5 (continued)

Required Action A.5 is modified by two Notes. Note 1 states that the QPT is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if Required Action A.5 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions. RAI 5 RAI 5

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time. RAI 5 RAI 5 RAI 5

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power. RAI 5 RAI 5

(continued)

BASES

ACTIONS (continued)

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1. RAI 5

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other indications and alarms available to the operator in the control room. For those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 24 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is > 75% RTP. RAI 5

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued)

Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 24 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. QAI 3

The symmetric thimble flux map can be used to measure symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map.

REFERENCES

1. 10 CFR 50.46.
2. FSAR Section 14.1.6.
3. FSAR Section 14.2.6.
4. FSAR Section 3.1.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO"

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
1-5	97	97	
3.10-4	103	103	
3.10-5	112	112	
3.10-8	181	181	
3.10-14	103	103	
3.10-15	112	112	

No changes to Rev 0.
Pages not included for Rev 1 review

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

Rev 1 change to DOC LA.2 only

QPTR can be maintained by monitoring excore and/or incore flux detectors.

This change is acceptable because ITS LCO 3.2.4 maintains the requirement that QPTR be maintained within required limits whenever thermal power is > 50% RTP. Moving requirements for the tilt alarm to the TRM is acceptable because a prompt change in quadrant power tilt (e.g., from a dropped rod) result in other indications of abnormality and is not usually identified by the tilt deviation monitor. Additionally, the 7 day frequency for ITS SR 3.2.4.1, the determination of QPTR, is adequate to detect any relatively slow changes in QPTR. Furthermore, requirements for the tilt deviation alarm will be maintained in the TRM which will require more frequent verification of QPTR if the deviation monitor is not functional.

The Quality Assurance Plan will be revised to specify that requirements in the TRM are part of the facility as described in the FSAR and that changes to the TRM can be made only in accordance with the requirements of 10 CFR 50.59. Therefore, this change is acceptable because there is no change to the existing requirements by the relocation of requirements to the TRM and future changes to the TRM will be controlled in accordance with 10 CFR 50.59.

LA.2 CTS 3.10.2.9 specifies that if one excore detector is out of service when operating above 75% RTP, then core quadrant power balances shall be determined using movable incore detectors using "at least two thimbles per quadrant." ITS SR 3.2.4.2 maintains the requirement in CTS 3.10.2.9 for the use of incore detectors to determine QPTR when operating > 75% RTP with an excore detector out of service. However, the stipulation that "at least two thimbles per quadrant" are needed to satisfy this requirement is relocated to the Bases for ITS SR 3.2.4.2. Additional clarification is added which states that the SR is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations.

This change is acceptable because ITS SR 3.2.4.2 maintains the requirement to use incore detectors when an excore detector is inoperable.

Maintaining the stipulation that "at least two thimbles per quadrant" are needed to satisfy this requirement in the ITS Bases is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight are maintained for the information being relocated out of the Technical Specifications.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 4:

No Significant Hazards Considerations
for
Changes between CTS and ITS
that are
Less Restrictive

No Significant Hazard Considerations for Changes that are Administrative, More Restrictive, and Removed Details are the same for all Packages. A Copy is included at the end of the Package.

No changes to Rev 0

Pages not included for Rev 1 review

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 5:

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

Status of NUREG 1431 Generic Changes for ITS 3.2.4

This ITS Specification is based on NUREG-1431 Specification No. **3.2.4**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-012	**025 R0**	REVISE ACTIONS TERMINOLOGY REGARDING QPTR TO MATCH ACTIONS BEING TAKEN	Rejected by NRC	Not Incorporated	**N/A**
WOG-045	**109 R0**	CLARIFY THE QPTR SURVEILLANCES	APPROVED/INCORPORATED	Incorporated	**T.3**
WOG-049	**110 R2**	DELETE SR FREQUENCIES BASED ON INOPERABLE ALARMS	APPROVED/INCORPORATED	Incorporated	**T.2**
WOG-059	**136 R0**	COMBINE LCO 3.1.1 AND 3.1.2	APPROVED/INCORPORATED	Incorporated	**T.1**
WOG-095	**241 R4**	ALLOW TIME FOR STABILIZATION AFTER REDUCING POWER DUE TO QPTR OUT OF LIMIT	Approved/Not in Orig	Incorporated	**T.4**
WOG-101		CLARIFY COMPLETION TIME AND FREQUENCY WORDING	TSTF Review	Not Incorporated	**N/A**

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

WOG-105	**314 R0**	REQUIRE STATIC AND TRANSIENT FQ MEASUREMENT	TSTF Review	Not Incorporated	**N/A**

<CTS> 3.2 POWER DISTRIBUTION LIMITS

<3.10.3> 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

<3.10.3.1> LCO 3.2.4 The QPTR shall be ≤ 1.02.

<3.10.3.1> <DOC A.3> <DOC A.4> APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP. R.1

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME	
<3.10.3.1> <3.10.3.1.a> <3.10.3.2> <DOC L.1>	A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination T.4	R.1
		AND			
<3.10.3.1.a> <DOC L.1> <DOC M.1>		A.2	Determine QPTR ~~Perform SR 3.2.4.1 and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > 1.00.~~	T.3 Once per 12 hours Insert: 3.2-18-01 T.4	R.1
		AND			
<DOC M.2>		A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours AND Once per 7 days thereafter	
		AND			
<DOC L.3>		A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1	
		AND			

(continued)

Typical

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

INSERT: 3.2-18-01

after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1 T.4 R.1

|R.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.5 --------NOTE--------- (S) (1.) Perform Required Action A.5 only after Required Action A.4 is completed. --------------------- ~~Calibrate excore detectors to show zero QPTR.~~	(T.4) Insert: 3.2-19-01 Insert: 3.2-19-02 R.1 Prior to increasing THERMAL POWER above the limit of Required Action A.1
<DOC L.3>	AND	
<DOC L.3>	A.6 --------NOTE--------- Perform Required Action A.6 only after Required Action A.5 is completed. --------------------- Perform SR 3.2.1.1 and SR 3.2.2.1.	Insert 3.2-19-03 ~~Within 24 hours after reaching RTP~~ ~~OR~~ ~~Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1~~ (T.4) R.1
<DOC A.11> <3.10.3.1.b> <3.10.3.2> B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

INSERT: 3.2-19-01

T.4

2. Required Action A.6 shall be completed whenever Required Action A.5 is performed.

INSERT: 3.2-19-02

T.4

Normalize excore detectors to restore QPTR to within limits.

INSERT: 3.2-19-03

T.4

Within 24 hours after achieving equilibrium conditions at RTP not to exceed 48 hours after increasing THERMAL POWER above the limit of Required Action A.1

T.4

≤ | R.1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
<DOC M.3> <DOC LA2> <1.11> <DOC A.9>	SR 3.2.4.1 -----------------NOTES----------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance ~~if adequate Power Range Neutron Flux channel inputs are not OPERABLE~~. --- Verify QPTR is within limit by calculation.	7 days ~~AND~~ ~~Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable~~	T.3 T.2 CLB.1
<3.10.2.9> <DOC A.10>	SR 3.2.4.2 Not -----------------NOTE----------------- ~~Only~~ required to be performed until 24 hours after if input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER ≥ 75% RTP. --- Verify QPTR is within limit using the movable incore detectors. > | T.4 R.1	Once within ~~12~~ 24 hours ~~AND~~ ~~12 hours thereafter~~	T.3 T.3 CLB.1 T.3

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made ~~during startup testing,~~ after refueling, and periodically during power operation.

Bank

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

6

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition; (Ref. 2)

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. ~~2~~ 3); and

Insert: B3.2-43-01

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. ~~3~~ 4).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

INSERT: B 3.2-43-01

225 calories/gram for non-irradiated fuel and 200 calories/gram for irradiated fuel

BASES

APPLICABLE SAFETY ANALYSES (continued)

Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36 ~~the NRC Policy Statement~~.

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient

(continued)

BASES

ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. Insert: B 3.2-45-01 T.4 R.1

A.2

After completion of Required Action A.1, the QPTR ~~alarm~~ may still ~~be in its alarmed state~~ exceed the specified limit. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. ~~If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly.~~ A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow. PA.1 T.2 R.1 T.4

Insert B 3.2-45-02 T.4 R.1

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

Insert B 3.2-45-03 T.4 R.1

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

INSERT: B 3.2-45-01

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of the QPTR determination, if necessary, to comply with the decreases maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to this revised limit.

INSERT: B 3.2-45-02

after achieving equilibrium conditions from a Thermal Power reduction per Required Action A.1.

INSERT: B 3.2-45-03

Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping.

BASES

ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

Insert B 3.2-46-01

A.5

Insert: B3.2-46-02

T.4 R.1

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are ~~recalibrated, to show a zero QPTR~~ prior to increasing THERMAL POWER to above the limit of Required Action A.1. This is done to detect any subsequent significant changes in QPTR.

two Notes. Note 1

restored to within limits

Required Action A.5 is modified by ~~a Note that~~ states that the QPT is not ~~zeroed out~~ until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). ~~This Note is~~ intended to prevent any ambiguity about the required sequence of actions.

Insert: B3.2-46-03

These Notes are

T.4 R.1

A.6

Once the flux tilt is ~~zeroed out~~ (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution ~~at RTP~~ is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the

R.1 T.4

insert: B 3.2-46-04

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

INSERT: B 3.2-46-01

normalized to restore QPTR to within limits

INSERT: B 3.2-46-02

Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00.

INSERT: B 3.2-46-03

Note 2 states that if Required Action A.2 is performed, then Required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6.

INSERT: B 3.2-46-04

achieving equilibrium conditions at RTP.

BASES

T.4 R.1

ACTIONS

A.6 (continued)

equilibrium conditions at

core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours ~~of the time when the ascent to power was begun~~. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time. T.4 R.1

Insert: B 3.2-47-01

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors ~~have been calibrated to show zero tilt~~ (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are ~~calibrated to show zero tilt~~ and the core returned to power. T.4 R.1

Insert B 3.2-47-02

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ~~≥~~ ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1 ~~if more than one input from Power Range Neutron Flux channels are inoperable~~. T.4 R.1 T.3

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

INSERT: B 3.2-47-01

after increasing THERMAL POWER above the limit of Required Action A.1

INSERT: B 3.2-47-02

normalized to restore QPTR to within limits

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

within its limits. The Frequency of 7 days ~~when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.~~

Insert: B 3.2-48-01

T.2

~~When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because~~ for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

until 24 hours after

T.3

This Surveillance is modified by a Note, which states that it is required ~~only when~~ the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ≥ 75% RTP.

not

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or ~~two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for three and four loop cores.~~

R.1

LA.2

Insert: B 3.2-48-02

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full

measure

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

T.2

INSERT: B 3.2-48-01

takes into account other indications and alarms available to the operator in the control room.

INSERT: B 3.2-48-02

two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. R.1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.2 (continued) incore monitoring of T.4 R.1

core flux map, to generate an incore QPTR. Therefore, QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. ~~Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.~~ PA 1

REFERENCES

1. 10 CFR 50.46.
2. ~~Regulatory Guide 1.77, Rev [0], May 1974.~~
3. ~~10 CFR 50, Appendix A, GDC 26.~~

Insert: B 3.2-49-01

INSERT: B 3.2-49-01

2. FSAR Section 14.1.6.

3. FSAR Section 14.2.6.

4. FSAR Section 3.1.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.2.4:
"QUADRANT POWER TILT RATIO (QPTR)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

RETENTION OF EXISTING REQUIREMENT (CURRENT LICENSING BASIS)

CLB.1 IP3 ITS differs from NUREG-1431 regarding the Frequency for performing QPTR determinations with incore detectors when an excore detector is inoperable. CTS 3.10.2.9 specifies that if one excore detector is out of service when > 75% RTP, then QPTR must be determined once a day (i.e., 24 hours) using incore detectors. Under the same conditions (one excore detector inoperable), NUREG-1431, Rev.1, SR 3.2.4.2 requires verification of QPTR using the incore detectors every 12 hours. (Note that SR 3.2.4.2 applies to one or more excore detectors inoperable while the CTS is limited to one excore detector inoperable. This difference is not significant because ITS LCO 3.3.1, Reactor Trip System (RTS) Instrumentation, would not allow continued operation with more than one inoperable excore detector.

Maintaining the current licensing basis is needed because IP3 experience indicates that using the incore probes every 12 hours creates a significant burden on plant operators and equipment without a commensurate increase in plant safety. Maintaining the CLB is acceptable because this option can only be used when 3 of the 4 excore detectors functional and the tilt deviation monitor is providing continuous indication of QPTR. Additionally, operation with an inoperable excore detector is expected to occur infrequently and then for only a limited period of time. Therefore, maintaining the CLB does not have a significant adverse impact on safety.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT

PA.1 Corrected typographical error or made a minor editorial improvement to improve clarity and ensure requirements are fully understood and consistently applied. There are no technical changes to requirements as specified in NUREG 1431, Rev. 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.2.4 - QUADRANT POWER TILT RATIO (QPTR)

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN OR DESIGN BASIS

DB.1 Design or implementation details are incorporated or revised as necessary to more precisely describe IP3 current design or practice. These changes are intended to describe the design, improve clarity, or ensure requirements are fully understood and consistently applied. Unless identified and described blow, these changes are self-explanatory. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-136 (WOG-59), Rev.1, which combines ITS 3.1.1,Shutdown Margin (SDM) – T_{avg} > 200°F, and ITS 3.1.2, Shutdown Margin (SDM) – T_{avg} ≤ 200°F, into ITS 3.1.1, Shutdown Margin (SDM). This change is necessary because ITS 3.1.1 and ITS 3.1.2 became essentially identical after Generic Change TSTF-09 (WOG-04.1), Rev.1, relocated values for shutdown margin during physics tests to COLR.

T.2 This change incorporates Generic Change TSTF-110, Rev.1 (WOG-49), which deletes the Frequency for SR 3.2.3.1 which requires verification of the Axial Flux Difference once within 1 hour and every 1 hour thereafter when the AFD monitor alarm is inoperable. These actions are relocated from the Technical Specifications to plant administrative practices because the alarms do not directly relate to the LCO limits (i.e., ITS LCO 3.2.3 still requires that AFD limits be maintained even if the alarm is not operable).

T.3 This change incorporates Generic Change TSTF-109, Rev.1 (WOG-45), which clarifies the requirements for performing SR 3.2.4.2 for calculating QPTR using the incore detectors. This change is needed because Required Action A.2 is intended to result in a periodic re-check and re-adjustment of thermal power based on QPTR. However, Required Action A.2 specifically requires performance of SR 3.2.4.1 which may not be

possible if Power Range Neutron Flux channel(s) are inoperable. In this event, SR 3.2.4.2 should be performed using the incore detectors. To more correctly specify the intended Required Action, A.2 is revised to simply require "Determine QPTR" rather than specifying an SR to perform. Additionally, Note 2 to SR 3.2.4.1 (QPTR by calculation) allows performance of SR 3.2.4.2 (QPTR using incore detectors) "if adequate Power Range Neutron Flux channel inputs are not OPERABLE." Besides posing some ambiguity as to what "adequate...inputs" are, it is overly restrictive. QPTR determination using incore detectors can adequately verify the requirements for QPTR in all cases; not just when flux channels are inoperable. SR 3.2.4.2 presentation of the frequency for verifying QPTR using incore detectors is revised to be consistent with typical presentation formats that provide for a period of time after establishing conditions.

T.4 This change incorporates Generic Change TSTF-241, Rev.4 (WOG-095), which clarifies Action completion times.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.1:
"Main Steam Safety Valves (MSSVs)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1 Reduce neutron flux trip setpoint to less than or equal to the applicable % RTP listed in Table 3.7.1-1.	4 hours
B. Required Action and associated Completion Time not met. OR One or more steam generators with less than two MSSVs OPERABLE.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------NOTE-------------------- Only required to be performed in MODES 1 and 2. --- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within $\pm 1\%$.	In accordance with the Inservice Testing Program

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Applicable Neutron Flux Trip Setpoint in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	APPLICABLE Neutron Flux Trip Setpoint (% RTP)
4	≤ 61
3	≤ 42
2	≤ 23

RAI-01

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER				LIFT SETTING (psig ± 3%)
STEAM GENERATOR				
#31	#32	#33	#34	
MS-45-1	MS-45-2	MS-45-3	MS-45-4	1065
MS-46-1	MS-46-2	MS-46-3	MS-45-4	1080
MS-47-1	MS-47-2	MS-47-3	MS-47-4	1095
MS-48-1	MS-48-2	MS-48-3	MS-48-4	1110
MS-49-1	MS-49-2	MS-49-3	MS-49-4	1120

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves and non-return valves, as described in the FSAR, Section 10.2 (Ref. 1). The five code safety valves per steam generator consist of four 6 inch by 10 inch and one 6 inch by 8 in. These valves are set to open at 1065, 1080, 1095, 1110 and 1120 psig, respectively. The steam generator safety valve capacity is rated to remove the maximum calculated steam flow (normally 105% of the maximum guaranteed steam flow) from the steam generators without exceeding 110% of the steam system design pressure, (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine or reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

(continued)

BASES

BACKGROUND (continued)

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 14 (Ref. 3). Of these, the full power loss of external electrical load without steam dump is the limiting AOO.

The transient response for loss of external electrical load without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. If a minimum reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT. The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems.

Startup and power operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by reducing the neutron flux trip setpoint and reducing THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. These limits on the neutron flux trip setpoint, specified in Table 3.7.1-1, are established based on guidance provided in Nuclear Safety Advisory Letter (NSAL) 94-001, Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves (Ref. 6) and Information Notice 94-60, Potential Overpressurization of Main Steam System (Ref. 7). The reactor trip setpoint reductions are calculated as follows:

$$Hi\phi = (100 / Q) [(wsh_{fg}N) / K]$$

Where:

RAI-01

(continued)

BASES

BACKGROUND (continued)

Hiϕ = Safety Analysis high neutron flux setpoint (% RTP);

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt (i.e.,3037 Mwt);

K = Conversion factor, 947.82 (Btu/sec)/Mwt;

ws = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure, including tolerance and accumulation, as appropriate, in lb/sec. (ws = 150 + 228.61 * (4 - V) lb/sec, where V = Number of inoperable safety valves in the steam line of the most limiting steam generator).

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm (i.e.,608.5 Btu/lbm).

N = Number of loops in plant (i.e., 4).

The calculated reactor trip setpoint is further reduced by 9% of full scale to account for instrument uncertainty and then rounded down.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36.

RAI-01

LCO

The accident analysis requires five MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that five MSSVs be OPERABLE in compliance with Reference 2. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 in the accompanying LCO, and Required Action A.1.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator

(continued)

BASES

LCO (continued)	overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program. The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure. This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.
APPLICABILITY	In MODE 1 above 23% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below 23% RTP in MODES 1, 2, and 3, only two MSSVs per steam generator are required to be OPERABLE. In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.
ACTIONS	The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV. <u>A.1</u> Startup and power operation with up to three of the five MSSVs associated with each steam generator inoperable is permissible if the maximum allowed power level is below the heat removing capability of the operable MSSVs. Therefore, startup and power operation with inoperable main steam line safety valves is allowable if the neutron flux trip setpoints are restricted within the limits specified in Table 3.7.1-1.

(continued)

BASES

ACTIONS

A.1 (continued)

This ensures that reactor power level is limited so that the heat input from the primary side will not exceed the heat removing capability of the OPERABLE MSSVs of the most limiting steam generator.

B.1 and B.2

RAI-01

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition.
3. FSAR, Section 14.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ANSI/ASME OM-1-1987.
6. Nuclear Safety Advisory Letter (NSAL) 94-001, Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves.
7. Information Notice 94-60, Potential Overpressurization of Main Steam System.

RAI-01

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.1:
"Main Steam Safety Valves (MSSVs)"

PART 5:

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

Status of NUREG 1431 Generic Changes for ITS 3.7.1

This ITS Specification is based on NUREG-1431 Specification No. **3.7.1**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-031		INCORPORATE A MSSV REQUIRED ACTION FOR UNITS LICENSED WITH A POSITIVE TEMPERATURE COEFFICIENT	Rejected by TSTF	Not Incorporated	N/A
WOG-083	235 R1	MSSV CHANGES	TSTF Review	Not Incorporated	N/A

3.7 PLANT SYSTEMS

<CTS>

3.7.1 Main Steam Safety Valves (MSSVs)

<3.4.A.1> LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

<3.4.A>
<DOC A.3> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----

<DOC A.7> Separate Condition entry is allowed for each MSSV.

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.4.A.1.a>	A. One or more required MSSVs inoperable. neutron flux trip setpoint	A.1	Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1.	4 hours (CLB.1)
<3.4.A.1.b> <DOC L.1>	B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
		AND		
	OR	B.2	Be in MODE 4.	12 hours
	One or more steam generators with less than [two] MSSVs OPERABLE.			

3.7.1-1
Typical

<CTS>
<Table 4.1-3 #4>
<DOC M.1>
<DOC L.2>
<DOC LA.1>
<DOC M.2>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.1.1	-------------------NOTE-------------------- Only required to be performed in MODES 1 and 2. --- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ±1%.	In accordance with the Inservice Testing Program

<CTS>

<T 3.4-1>

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Applicable ~~Power~~ in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	APPLICABLE POWER (% RTP)
~~5~~	~~≤ 100~~
4	≤ 80 61
3	≤ 60 42
2	≤ 40 23

Neutron Flux Trip Setpoint

CLB.1

R.1

<CTS>
<3.4.A.1>
<DOC M1>

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER				LIFT SETTING (psig ± ~~[~~3~~]~~%)
STEAM GENERATOR				
#31 ~~#1~~	#32 ~~#2~~	~~[#3]~~ #33	#34 ~~[#4]~~	
MS-45-1	MS-45-2	MS-45-3	MS-45-4	1065
MS-46-1	MS-46-2	MS-46-3	MS-46-4	1080
MS-47-1	MS-47-2	MS-47-3	MS-47-4	1095
MS-48-1	MS-48-2	MS-48-3	MS-48-4	1110
MS-49-1	MS-49-2	MS-49-3	MS-49-4	1120

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

and non-return valves

DB.1

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section ~~[10.3.1]~~ 10.2 (Ref. 1). ~~The MSSV capacity criteria is 110% of rated steam flow at 110% of the steam generator design pressure. This meets the requirements of the ASME Code, Section III~~ (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine or reactor trip.

Insert: B 3.7-1-01

PA.1

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section ~~[15.2]~~ 14 (Ref. 3). Of these, the full power ~~turbine trip~~ loss of external electrical load without steam dump is the limiting AOO. ~~This event also terminates normal feedwater flow to the steam generators.~~

DB.1

The transient response for ~~turbine trip~~ loss of external electrical load without a direct reactor trip presents no hazard to the integrity of the RCS

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.1 - Main Steam Safety Valves (MSSVs)

INSERT: B 3.7-1-01

The five code safety valves per steam generator consist of four 6 inch by 10 inch and one 6 inch by 8 in. These valves are set to open at 1065, 1080, 1095, 1110 and 1120 psig, respectively. The steam generator safety valve capacity is rated to remove the maximum calculated steam flow (normally 105% of the maximum guaranteed steam flow) from the steam generators without exceeding 110% of the steam system design pressure.

BASES

APPLICABLE SAFETY ANALYSES (continued)

or the Main Steam System. If a minimum reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT. The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems. ~~The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.~~ DB.2 R.1

Insert: B 3.7-2-01

The MSSVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

LCO

The accident analysis requires ~~four~~ five MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that five MSSVs be OPERABLE in compliance with Reference 2, ~~even though this is not a requirement of the DBA analysis.~~ This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 in the accompanying LCO, and Required Action ~~A.2~~ A.1.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.1 - Main Steam Safety Valves (MSSVs)

INSERT: B 3.7-2-01

Startup and power operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by reducing the neutron flux trip setpoint and reducing THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. These limits on the neutron flux trip setpoint, specified in Table 3.7.1-1, are established based on guidance provided in Nuclear Safety Advisory Letter (NSAL) 94-001, Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves (Ref. 6) and Information Notice 94-60, Potential Overpressurization of Main Steam System (Ref. 7). The reactor trip setpoint reductions are calculated as follows:

$$Hi\phi = (100 / Q)\ [(ws h_{fg} N) / K]$$

Where:

$Hi\phi$	=	Safety Analysis high neutron flux setpoint (% RTP);
Q	=	Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt (i.e.,3037 Mwt);
K	=	Conversion factor, 947.82 (Btu/sec)/Mwt;
ws	=	Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSVopening pressure, including tolerance and accumulation, as appropriate, in lb/sec. (ws = 150 + 228.61 * (4 - V) lb/sec, where V = Number of inoperable safety valves in the steam line of the most limiting steam generator).
h_{fg}	=	Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm (i.e.,608.5 Btu/lbm).
N	=	Number of loops in plant (i.e., 4).

The calculated reactor trip setpoint is further reduced by 9% of full scale to account for instrument uncertainty and then rounded down.

BASES

LCO (continued)

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

23%

In MODE 1 above 40% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below 40% RTP in MODES 1, 2, and 3, only two MSSVs per steam generator are required to be OPERABLE.

CLB.1

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

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

A.1

With one or more MSSVs inoperable, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 20%. This is accomplished by reducing THERMAL POWER by at least 20%, which conservatively limits the energy transfer to all steam generators to approximately 80% of total capacity, consistent with the relief capacity of the most limiting steam generator.

Insert B 3.7-3-01

R.1

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.1 - Main Steam Safety Valves (MSSVs)

INSERT: B 3.7-3-01

Startup and power operation with up to three of the five MSSVs associated with each steam generator inoperable is permissible if the maximum allowed power level is below the heat removing capability of the operable MSSVs. Therefore, startup and power operation with inoperable main steam line safety valves is allowable if the neutron flux trip setpoints are restricted within the limits specified in Table 3.7.1-1. This ensures that reactor power level is limited so that the heat input from the primary side will not exceed the heat removing capability of the OPERABLE MSSVs of the most limiting steam generator.

BASES

ACTIONS

A.1 (continued)

CLB.1

For each steam generator, at a specified pressure, the fractional relief capacity (FRC) of each MSSV is determined as follows:

$$FRC = \frac{A}{B}$$

where:

A = the relief capacity of the MSSV; and

B = the total relief capacity of all the MSSVs of the steam generator.

The FRC is the relief capacity necessary to address operation with reduced THERMAL POWER.

The reduced THERMAL POWER levels in the LCO prevent operation at power levels greater than the relief capacity of the remaining MSSVs. The reduced THERMAL POWER is determined as follows:

$$RP = \left[1 - (N_1 \times FRC_1 + N_2 \times FRC_2 + \cdots + N_5 \times FRC_5)\right] \times 100\%$$

where:

RP = Reduced THERMAL POWER for the most limiting steam generator expressed as a percent of RTP;

N_1, N_2, ..., N_5 represent the status of the MSSV 1, 2, ..., 5, respectively,

= 0 if the MSSV is OPERABLE,
= 1 if the MSSV is inoperable;

FRC_1, FRC_2, ..., FRC_5 = the relief capacity of the MSSV 1, 2, ..., 5, respectively, as defined above.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria ~~, and~~ .

~~e. Verification of the balancing device integrity on balanced valves.~~ DB.1

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± [3]% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section ~~10.3.1~~. 10.2
2. ASME, Boiler and Pressure Vessel Code, Section III, ~~Article NC-7000, Class 2 Components~~. 1971 Edition
3. FSAR, Section ~~15.2~~. 14
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ANSI/ASME OM-1-1987.

Insert B 3.7-6-01

R1

6. Nuclear Safety Advisory Letter (NSAL) 94-001, Operation at Reduced Power Levels with Inoperable Main Steam Safety Valves.

7. Information Notice 94-60, Potential Overpressurization of Main Steam System.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.2:

"Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

BASES

APPLICABLE SAFETY ANALYSES (continued)

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36.

LCO

This LCO requires that four MSIVs and four MSCVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. The MSCVs are considered OPERABLE when inspections and testing required by the Inservice Test Program are completed at the specified FREQUENCY in accordance with SR 3.7.2.2.

RAI-02

This LCO provides assurance that the MSIVs and MSCVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs and MSCVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when MSIVs are closed. These are the conditions when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low and the potential for and consequences of an SLB are significantly reduced.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

Each MSCV must be inspected to ensure that it closes properly. This ensures that the safety analysis assumptions are met. The Frequency of this SR is based on Inservice Testing Program requirements and corresponds to the expected refueling cycle.

RAI-0<

REFERENCES

1. FSAR, Section 10.2.
2. FSAR, Section 6.
3. FSAR, Section 14.
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.2:

"Main Steam Isolation Valves (MSIVs) and Main Steam Check Valves (MSCVs)"

PART 5:

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

Status of NUREG 1431 Generic Changes for ITS 3.7.2

This ITS Specification is based on NUREG-1431 Specification No. **3.7.2**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-032	**102 R0**	EXTEND THE PERIODIC VERIFICATION OF INOPERABLE MSIV AND MFIV CLOSURE TO 31 DAYS	Rejected by NRC	Not Incorporated	**N/A**
WOG-064	**281 R0**	MSIV AOT TO 72 HOURS	TSTF Review	Not Incorporated	**N/A**
WOG-098	**289 R0**	SEPARATE CLOSURE TIME TESTING AND ACTUATION SIGNAL TESTING FOR MSIVS AND FWIVS	APPROVED/NOT INCORP	Not Incorporated	**N/A**

BASES

This case

APPLICABLE SAFETY ANALYSES (continued)

b. A break outside of containment and upstream from the MSIVs. is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

c. A break downstream of the MSIVs. will be isolated by the closure of the MSIVs. This case

d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases. In this case

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36

LCO

This LCO requires that ~~[four]~~ MSIVs and four MSCVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal. Insert: B3.7-9-01

This LCO provides assurance that the MSIVs and MSCVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

Insert: B 3.7-9-02

APPLICABILITY

The MSIVs and MSCVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when MSIVs are closed ~~and de-activated~~ when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, ~~normally most of the MSIVs are closed~~ and the steam generator energy is low. Insert: B 3.7-9-03

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.2 - Main Steam Isolation Valves (MSIVs)
and Main Steam Check Valves (MSCVs)

INSERT: B 3.7-9-01

The MSCVs are considered OPERABLE when inspections and testing required by the Inservice Test Program are completed at the specified FREQUENCY in accordance with SR 3.7.2.2.

INSERT: B 3.7-9-02

These are the conditions

INSERT: B 3.7-9-03

and the potential for and consequences of an SLB are significantly reduced.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

The Frequency is in accordance with the [Inservice Testing Program ~~or [18] months~~]. The [18] month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

this

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5. ~~exercising requirements.~~ This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

Insert: B 3.7-12-01

REFERENCES

1. FSAR, Section [10.3]. 10.2
2. FSAR, Section [6.2]. 6
3. FSAR, Section [15.1.5]. 14
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

NUREG-1431 Markup Inserts
ITS SECTION 3.7.2 - Main Steam Isolation Valves (MSIVs)
and Main Steam Check Valves (MSCVs)

INSERT: B 3.7-12-01

SR 3.7.2.2

Each MSCV must be inspected to ensure that it closes properly. This ensures that the safety analysis assumptions are met. The Frequency of this SR is based on Inservice Testing Program requirements and corresponds to the expected refueling cycle.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.3:

"Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (MBFRVs) and MBFRV Low Flow Bypass Valves"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.7 PLANT SYSTEMS

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

LCO 3.7.3 Two MBFPDVs, four MFRVs and four MFRV low flow bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when MBFPDVs, or MFRVs and MFRV low flow bypass valves are closed and de-activated or isolated by a closed manual valve.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or both MBFPDVs inoperable.	A.1	Close or isolate MBFPDV.	72 hours
	<u>AND</u>		
	A.2	Verify MBFPDV is closed or isolated.	Once per 7 days
B. One or more MFRVs inoperable.	B.1	Close or isolate MFRV.	72 hours
	<u>AND</u>		
	B.2	Verify MFRV is closed or isolated.	Once per 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more MFRV low flow bypass valves inoperable.	C.1	Close or isolate bypass valve.	72 hours
	<u>AND</u>		
	C.2	Verify bypass valve is closed or isolated.	Once per 7 days
D. Two valves in series in the same flow path inoperable.	D.1	Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify each MBFPDV, MFRV and MFRV low flow bypass valve closes on an actual or simulated actuation signal within the following limits: a. MBFPDV closure time ≤ 122 seconds; b. MFRV closure time ≤ 10 seconds; and, c. MFRV Low Flow Bypass valve closure time ≤ 10 seconds.	In accordance with the Inservice Testing Program

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (MFRVs) and MFRV Low Flow Bypass Valves

BASES

BACKGROUND

The MBFPDVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. Closure of the two MBFPDVs or four MFRVs and four MFRV low flow bypass valves terminates flow to the steam generators. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MBFPDVs will be mitigated by their closure. Closure of the MBFPDVs or MFRVs and MFRV low flow bypass valves, effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

In the event of a secondary side pipe rupture inside containment, either the MBFPDVs or MFRVs and MFRV low flow bypass valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MBFPDV is located on the discharge of each of the two Main Boiler Feedpumps (MBFPs), and one MFRV and MFRV low flow bypass valve, is located on each of the four MFW lines, outside but close to containment. The MFIVs and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MBFPDV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

(continued)

BASES

BACKGROUND (continued)

The two MBFPDVs, four MFRVs and four MFRV low flow bypass valves will close on receipt of an ESFAS Safety Injection signal. An ESFAS Tavg – Low coincident with reactor trip will close the four MFRVs and four MFRV low flow bypass valves. A Steam Generator Hi-Hi level trip will close the MBFPDV and MFRVs and MFRV low flow bypass valves associated with the affected SG. They may also be closed manually. In addition to the two MBFPDVs, four MFRVs and four MFRV low flow bypass valves, a check valve outside containment is available. The check valve isolates the feedwater line to prevent blowdown of a SG if main or auxiliary feedwater pressure are lost.

A description of the MBFPDVs and MFRVs is found in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MBFPDVs and MFRVs is established by the analyses for the large SLB. Closure of the MBFPDVs, MFRVs and MFRV low flow bypass valves, may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level – high high or a feedwater isolation signal. Feedwater isolation also occurs as a result of any safety injection signal.
Failure of an MBFPDV in conjunction with the failure of an MFRV or MFRV low flow bypass valve to close following an SLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MBFPDVs, MFRVs and MFRVs Low Flow Bypass Valves satisfy Criterion 3 of 10 CFR 50.36.

RAI-01

LCO

This LCO ensures that the MBFPDVs, MFRVs and MFRV low flow bypass valves will isolate MFW flow to the steam generators, following a main steam line break.

(continued)

BASES

LCO (continued)

This LCO requires that two MBFPDVs, four MFRVs and four MFRV low flow bypass valves be OPERABLE. The MBFPDVs, MFRVs and MFRV low flow bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A feedwater isolation signal on a steam generator water level – high high signal and this function is relied on to terminate an excess feedwater flow event; therefore, failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MBFPDVs, MFRVs and MFRV bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MBFPDVs, MFRVs and MFRV bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function. A de-activated motor operated valve is considered to be a manual valve.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MBFPDVs, MFRVs and MFRV bypass valves are normally closed since MFW is not required.

ACTIONS

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

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

With one MFPDV in one or both flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves, the MBFP trip function, and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on industry operating experience.

Inoperable MBFPDVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on industry operating experience.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of other administrative controls to ensure that the valves are closed or isolated.

C.1 and C.2

With one MFRV low flow bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on industry operating experience.

Inoperable associated bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of the administrative controls that ensure that these valves are closed or isolated.

D.1

With two inoperable valves in series in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action

(continued)

BASES

ACTIONS

D.1 (continued)

returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MBFPDV or MFRV, or otherwise isolate the affected flow path.

E.1 and E.2

If the MBFPDV(s), MFRV(s), and MFRV bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MBFPDV(s), MFRV(s), and MFRV bypass valves is within required limits on an actual or simulated actuation signal. The closure times are assumed in the accident and containment analyses. The acceptance criteria for this SR do not include the 2 second delay associated with the ESFAS activation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves can not be tested at power because valve closure or even a part stroke exercise increases the risk of a valve closure and MBFP trip. This is consistent with the ASME Code, Section XI (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1 (continued)

The Frequency for this SR is in accordance with the Inservice Testing Program. The required Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the required Frequency.

REFERENCES

1. FSAR, Section 10.2.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.3:

"Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (MFRVs) and MFRV Low Flow Bypass Valves"

PART 5:

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

Status of NUREG 1431 Generic Changes for ITS 3.7.3

This ITS Specification is based on NUREG-1431 Specification No. **3.7.3**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-032	**102 R0**	EXTEND THE PERIODIC VERIFICATION OF INOPERABLE MSIV AND MFIV CLOSURE TO 31 DAYS	Rejected by NRC	Not Incorporated	**N/A**
WOG-098	**289 R0**	SEPARATE CLOSURE TIME TESTING AND ACTUATION SIGNAL TESTING FOR MSIVS AND FWIVS	APPROVED/NOT INCORP	Not Incorporated	**N/A**

3.7 PLANT SYSTEMS

Insert: 3.7-7-02

3.7.3 ~~Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) [and Associated Bypass Valves]~~ DB.1

LCO 3.7.3 ~~[Four] MFIVs, [four] MFRVs, [and associated bypass valves]~~ shall be OPERABLE.

Insert: 3.7-7-03

Insert: 3.7-7-04

<DOC A.3> APPLICABILITY: MODES 1, 2, [and 3] except when ~~MFIV, MFRV, [or associated bypass valve]~~ ~~is~~ are closed and [de-activated] [or isolated by a closed manual valve].

ACTIONS

-----NOTE-----
<DOC M.1> Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.1> A. One or ~~more~~ both MFIVs MBFPDVs inoperable.	A.1 Close or isolate MFIV MBFPDV. AND A.2 Verify MFIV MBFPDV is closed or isolated.	[72] hours Once per 7 days
<DOC M.1> B. One or more MFRVs MBFRVs inoperable.	B.1 Close or isolate MFRV MBFRV. AND B.2 Verify MFRV is closed or isolated.	[72] hours Once per 7 days

(continued)

NUREG-1431 Markup Inserts

ITS SECTION 3.7.3 - Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (M~~B~~FRVs) and M~~B~~FRV Bypass Valves

R.1
typical

INSERT: 3.7-7-01

MBFPDVs, M~~B~~FRVs and M~~B~~FRV Low Flow Bypass Valves

INSERT: 3.7-7-02

Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (M~~B~~FRVs) and M~~B~~FRV Low Flow Bypass Valves

INSERT: 3.7-7-03

Two MBFPDVs, four M~~B~~FRVs and four M~~B~~FRV low flow bypass valves

INSERT: 3.7-7-04

MBFPDVs or M~~B~~FRV and M~~B~~FRV low flow

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<DOC M.1>	[C. One or more ~~MFRV or preheater~~ bypass valves inoperable. MFRV low flow	C.1	Close or isolate bypass valve.	[72] hours
		AND		
		C.2	Verify bypass valve is closed or isolated.]	Once per 7 days
<DOC M.1> <DOC A.3>	D. Two valves in series in the same flow path inoperable.	D.1	Isolate affected flow path.	8 hours
<DOC M.1>	E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
		AND		
		E.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify ~~the closure time of each MFIV, MFRV [, and associated bypass valve] is ≤ [7] seconds~~ Insert: 3.7-8-01 on an actual or simulated actuation signal. Insert 3.7-8-02	In accordance with the [Inservice Testing Program ~~or [18] months~~

NUREG-1431 Markup Inserts

ITS SECTION 3.7.3 - Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (M~~B~~FRVs) and M~~B~~FRV Bypass Valves

INSERT: 3.7-8-01

each MBFPDV, M~~B~~FRV and M~~B~~FRV low flow bypass valve closes

INSERT: 3.7-8-02

within the following limits:

a. MBFPDV closure time ≤ 122 seconds;

b. M~~B~~FRV closure time ≤ 10 seconds; and,

c. M~~B~~FRV Low Flow Bypass valve closure time ≤ 10 seconds.

B 3.7 PLANT SYSTEMS Insert: B 3.7-13-02

B 3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) ~~[and Associated Bypass Valves]~~

BASES

BACKGROUND

MBFPDVs

MØFRVs

Insert: B 3.7-13-03

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. Closure of the MFIVs and associated ~~bypass valves or MFRVs and associated bypass valves~~ terminates flow to the steam generators. ~~terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs.~~ The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs ~~and associated bypass valves~~, or ~~MFRVs and associated~~ bypass valves, effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

MBFPDVs

MØFRVs and MØFRV low flow

~~The MFIVs and associated bypass valves, or MFRVs and associated bypass valves, isolate the nonsafety related portions from the safety related portions of the system.~~ In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

Insert: B 3.7-13-04

Insert B 3.7-13-05

One ~~MFIV and associated bypass valve~~, as and one of the four ~~MFRV and its associated~~ bypass valve, ~~are~~ is located on each MFW line, outside but close to containment. The MFIVs and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following ~~MFIV or MFRV~~ closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

MØFRV and MØFRV low flow

MBFPDVs and MØFRVs

MBFPDV or MØFRV

Insert B 3.7-13-06

~~The MFIVs and associated bypass valves, and MFRVs and associated bypass valves, close on receipt of a T_{avg}—Low coincident with reactor trip (P-4) or steam generator water~~

~~(continued)~~

R.1

NUREG-1431 Markup Inserts

ITS SECTION 3.7.3 - Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (M~~B~~FRVs) and M~~B~~FRV Bypass Valves

INSERT: B 3.7-13-01

MBFPDVs, M~~B~~FRVs and M~~B~~FRV Low Flow Bypass Valves

INSERT: B 3.7-13-02

Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (M~~B~~FRVs) and M~~B~~FRV Low Flow Bypass Valves

INSERT: B 3.7-13-03

two MBFPDVs or four M~~B~~FRVs and four M~~B~~FRV low flow bypass valves

INSERT: B 3.7-13-04

either the MBFPDVs or M~~B~~FRVs and M~~B~~FRV low flow bypass valves

INSERT: B 3.7-13-05

MBFPDV is located on the discharge of each of the two Main Boiler Feedpumps (MBFPs)

INSERT: B 3.7-13-06

The two MBFPDVs, four M~~B~~FRVs and four M~~B~~FRV low flow bypass valves will close on receipt of an ESFAS Safety Injection signal. An ESFAS Tavg—Low coincident with reactor trip will close the four M~~B~~FRVs and four M~~B~~FRV low flow bypass valves. A Steam Generator Hi-Hi level trip will close the MBFPDV and M~~B~~FRVs and M~~B~~FRV low flow bypass valves associated with the affected SG.

BASES

BACKGROUND (continued)

~~level—high high signal~~. They ~~may~~ also be ~~actuated~~ closed manually. In addition to the ~~MFIVs and associated bypass valves, and the MFRVs and associated bypass valves~~, a check valve ~~inside~~ outside containment is available. The check valve isolates the feedwater line~~, penetrating containment, and ensures that the consequences of events do not exceed the capacity of the containment heat removal systems~~.

Insert: B 3.7-14-01

Insert: B 3.7-14-02

A description of the ~~MFIVs and MFRVs~~ MBFPDVs and MØFRVs is found in the FSAR, Section ~~10.4.7~~ 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the ~~MFIVs and MFRVs~~ MBFPDVs and MØFRVs is established by the analyses for the large SLB. ~~It is also influenced by the accident analysis for the large FWLB~~. Closure of the ~~MFIVs and associated bypass valves, or MFRVs and associated bypass valves~~ may also be relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level—high high ~~signal~~ or a feedwater isolation signal ~~on high steam generator level~~.

Insert: B 3.7-14-03

Insert: B 3.7-14-04

Insert: B. 3.7-14-05

Failure of ~~an MFIV, MFRV, or the associated bypass valves~~ to close following an SLB ~~or FWLB~~ can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The ~~MFIVs and MFRVs~~ MBFPDVs, MØFRVs and MØFRU low flow bypass valves satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36.

R

LCO

This LCO ensures that the ~~MFIVs, MFRVs, and their associated bypass valves~~ will isolate MFW flow to the steam generators, following ~~an FWLB or~~ main steam line break. ~~These valves will also isolate the nonsafety related portions from the safety related portions of the system~~. (a)

Insert: B3.7-14-06

This LCO requires that ~~[four] MFIVs and associated bypass valves and [four] MFRVs [and associated bypass valves] be OPERABLE. The MFIVs and MFRVs and the associated bypass valves~~ are considered OPERABLE when isolation times are

Insert: B3.7-14-07

~~(continued)~~

NUREG-1431 Markup Inserts

ITS SECTION 3.7.3 - Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (M~~B~~FRVs) and M~~B~~FRV Bypass Valves

INSERT: B 3.7-14-01

two MBFPDVs, four M~~B~~FRVs and four M~~B~~FRV low flow bypass valves

INSERT: B 3.7-14-02

to prevent blowdown of a SG if main or auxiliary feedwater pressure are lost.

INSERT: B 3.7-14-03

MBFPDVs, M~~B~~FRVs and M~~B~~FRV low flow bypass valves

INSERT: B 3.7-14-04

Feedwater isolation also occurs as a result of any safety injection signal.

INSERT: B 3.7-14-05

an MBFPDV in conjunction with the failure of an M~~B~~FRV or M~~B~~FRV low flow bypass valve

INSERT: B 3.7-14-06

MBFPDVs, M~~B~~FRVs and M~~B~~FRV low flow bypass valves

INSERT: B 3.7-14-07

two MBFPDVs, four M~~B~~FRVs and four M~~B~~FRV low flow bypass valves be OPERABLE. The MBFPDVs, M~~B~~FRVs and M~~B~~FRV low flow bypass valves

BASES

LCO (continued)

within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. (A) ~~If~~ a feedwater isolation signal on ~~high steam generator level~~ is relied on to terminate an excess feedwater flow event; Therefore, failure to meet the LCO may result in the introduction of water into the main steam lines.

Insert: B 3.7-15-01

APPLICABILITY

Insert B 3.7-15-02

The ~~MFIVs and MFRVs and the associated~~ bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, [and 3], the ~~MFIVs and MFRVs and the associated~~ bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function.

Insert B 3.7-15-03

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the ~~MFIVs, MFRVs, and the associated~~ bypass valves are normally closed since MFW is not required.

ACTIONS

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

A.1 and A.2

With one ~~MFIV~~ MFPDV in one or ~~more~~ both flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within [72] hours. When these valves are closed or isolated, they are performing their required safety function.

~~(continued)~~

NUREG-1431 Markup Inserts
ITS SECTION 3.7.3 - Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (M~~B~~FRVs) and M~~B~~FRV Bypass Valves

INSERT: B 3.7-15-01

a steam generator water level – high high signal and this function

INSERT: B 3.7-15-02

MBFPDVs, M~~B~~FRVs and M~~B~~FRV

INSERT: B 3.7-15-03

A de-activated motor operated valve is considered to be a manual valve.

BASES

ACTIONS

A.1 and A.2 (continued)

Insert B 3.7-16-01

The {72} hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The {72} hour Completion Time is reasonable, based on industry operating experience.

MBFPDVs

Inoperable MFIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

MØFRV

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within {72} hours. When these valves are closed or isolated, they are performing their required safety function.

The [72] hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The {72} hour Completion Time is reasonable, based on industry operating experience.

MØFRVs

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of ~~valve status indications available in the control room, and~~ other administrative controls to ensure that the valves are closed or isolated.

~~(continued)~~

INSERT: B 3.7-16-01

, the MBFP trip function,

BASES

ACTIONS (continued)

C.1 and C.2

MØFRV low flow

With one ~~associated~~ bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within ~~[~~72~~]~~ hours. When these valves are closed or isolated, they are performing their required safety function.

The ~~[~~72~~]~~ hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The ~~[~~72~~]~~ hour Completion Time is reasonable, based on industry operating experience.

Inoperable associated bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of the ~~valve status indications available in the control room, and other~~ administrative controls ~~to~~ that ensure that these valves are closed or isolated.

D.1

series in

With two inoperable valves in the same flow path, there may be no redundant system to operate automatically and perform the required safety function. ~~Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path.~~ Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the ~~MFIV or MFRV~~, or otherwise isolate the affected flow path.

MBFPDV or MØFRV

~~(continued)~~

BASES

ACTIONS (continued)

E.1 and E.2

Insert: B3.7-18-01

If the ~~MFIV(s) and MFRV(s) and the associated~~ bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, [and in MODE 4 within 12 hours]. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

Insert: B3.7-18-04

Insert: B3.7-18-02

Insert: B3.7-18-03

This SR verifies that the closure time of each ~~MFIV, MFRV, and associated~~ bypass valves is ~~≤ 7 seconds~~ on an actual or simulated actuation signal. The ~~MFIV and MFRV~~ closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves ~~should~~ can not be tested at power ~~since~~ because valve closure on even a part stroke exercise increases the risk of a valve closure ~~with the unit generating power~~ and MBFP trip. This is consistent with the ASME Code, Section XI (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the [Inservice Testing Program ~~or [18] months~~]. The ~~[18] month~~ required Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~[18] month~~ required Frequency.

REFERENCES

1. FSAR, Section ~~[10.4.7]~~ 10.2.
2. ASME, Boiler and Pressure Vessel Code, Section XI.

NUREG-1431 Markup Inserts

ITS SECTION 3.7.3 - Main Boiler Feedpump Discharge Valves (MBFPDVs), Main Feedwater Regulation Valves (M~~B~~FRVs) and M~~B~~FRV Bypass Valves

INSERT: B 3.7-18-01

MBFPDV(s), M~~B~~FRV(s) and M~~B~~FRV

INSERT: B 3.7-18-02

MBFPDV(s), M~~B~~FRV(s) and M~~B~~FRV

INSERT: B 3.7-18-03

within required limits

INSERT: B 3.7-18-04

The acceptance criteria for this SR do not include the 2 second delay associated with the ESFAS actuation signal.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.5: "Auxiliary Feedwater (AFW) System"

PART 1:

Indian Point 3 Improved Technical Specifications and Bases

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

------------------------------NOTE------------------------------
Only one AFW train, which includes a motor driven pump capable of supporting the credited steam generator(s), is required to be OPERABLE in MODE 4.

RAI 3

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met. OR Two AFW trains inoperable in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 18 hours
D. Three AFW trains inoperable in MODE 1, 2, or 3.	------------NOTE------------ LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------------------------- D.1 Initiate action to restore one AFW train to OPERABLE status.	 Immediately
E. Required AFW train inoperable in MODE 4.	E.1 Initiate action to restore AFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	-------------------NOTE------------------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. -- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	-------------------NOTE------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. -- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with Inservice Testing Program
SR 3.7.5.3	-------------------NOTE------------------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. -- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.5.4	---NOTES--- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. --- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	24 months

RAI-01 (typo)

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction from the condensate storage tank (CST) (LCO 3.7.6) and pump to the steam generator secondary side via a connection to the main feedwater (MFW) piping at a point outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the steam bypass (High Pressure Steam Dump) valves and recirculated to the CST. RAI-02

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. FSAR Section 10.2 (Ref. 1) describes this configuration as two pumping loops using two different types of motive power to the pumps. One auxiliary feedwater loop utilizes a steam turbine driven pump and the other utilizes two motor driven pumps. Technical specifications describe this configuration as three trains because each motor driven pump provides 100% of AFW flow capacity, and, depending on steam conditions, the turbine driven pump capacity approaches 200% of the required capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent power supply and feeds two steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

(continued)

BASES

BACKGROUND (continued)

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The turbine driven AFW pump supplies a common header capable of feeding all steam generators. Each of the steam generators can also be supplied by one of the two motor driven AFW pumps. Any of the three pumps at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The motor driven pumps are actuated by any one of the following:

1) Low-low level in any steam generator;

2) Loss of voltage (Non SI blackout) on 480 VAC bus 2A/3A (starts AFW Pump 31) and loss of voltage (Non SI blackout) on 480 VAC bus 6A (starts AFW Pump 33);

3) Safety Injection signal;

4) Auto trip of either main boiler feed pump;

5) Manual actuation from the Control Room; and

6) Manual actuation locally at the pump room.

The steam turbine driven pump is actuated by any one of the following:

1) Low-low level in two of the four steam generators;

2) Loss of voltage (Non SI blackout) on 480 VAC busses 2A/3A or 6A;

(continued)

BASES

BACKGROUND (continued)

3) Manual actuation from the Control Room; and

4) Manual actuation locally at the pump room.

The steam driven AFW pump must be throttled manually in order to bring the unit up to speed after a start signal. In addition, the steam driven pump discharge flow control valves must be manually opened as necessary to provide adequate auxiliary feedwater flow.

The AFW System is discussed in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus accumulation.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW System flow must also be available to account for flow losses such as pump recirculation and line breaks.

The limiting events that require the AFW System are as follows:

a. small break loss of coolant accident;

b. loss of AC sources; and

c. loss of feedwater.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The AFW turbine driven pump actuates automatically when required to ensure an adequate feedwater supply to the steam generators is available during loss of power. Power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36.

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of events that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps are required to be OPERABLE to ensure the capability to maintain the plant in hot shutdown with a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

The AFW System is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE, each supplying AFW to two separate steam generators. The turbine driven AFW pump is required to be OPERABLE with steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to all of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. The motor driven AFW pump required to be OPERABLE in Mode 4 must be capable of supporting the SG(s) being credited as the redundant decay heat removal path in accordance with LCO 3.4.6, RCS Loops - MODE 4. This requirement ensures the ability to

RAI-3

(continued)

BASES

LCO (continued)	maintain the required level in the SG(s) (and decay heat removal capacity) during extended periods in Mode 4 with or without offsite power. Requiring only one OPERABLE AFW pump is acceptable because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump. RAI 3
APPLICABILITY	In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory needed to achieve and maintain MODE 4 conditions. In MODE 4, a motor driven AFW pump may be needed to support heat removal via the steam generators. In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.
ACTIONS	A.1 If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons: a. The redundant OPERABLE steam supply to the turbine driven AFW pump; b. The availability of redundant OPERABLE motor driven AFW pumps; and c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

(continued)

BASES

ACTIONS

A.1 (continued)

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

(continued)

BASES

ACTIONS (continued)

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note that states the SR is not required in MODE 4. Not performing this SR in MODE 4 is acceptable for the following reasons: AFW pumps are typically operated intermittently to keep the SGs filled when in MODE 4, the decay heat load is low; an RHR loop is required to be OPERABLE as the primary method of decay heat removal in Mode 4; and, the SG is required to be maintained at a level that ensures a significant inventory is available as a heat sink before the AFW pump is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

required to refill the SG. These factors ensure that a significant amount of time would be available to complete any valve realignments needed to refill a SG when in Mode 4.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test when SG pressure is < 600 psig.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.3 (continued)

required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is operated as necessary to maintain SG water level.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is operated as necessary and the autostart function is not required. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral allows the test to be performed at rated conditions. Note 2 states that the SR is not required in MODE 4. In MODE 4, the required pump is operated as necessary to maintain SG water level and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 10.2.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.5:
"AUXILIARY FEEDWATER SYSTEM"

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
3.4-1	92	92	
3.4-2	92	92	
3.4-3	151	151	
4.8-1	178;98-043	185	No impact
4.8-2	0	0	

Rev 1 correction to CTS markup page 4.8-1

Add SR 3.7.5.1 M.4

ITS 3.7.5
A.1

4.8 AUXILIARY FEEDWATER SYSTEM

~~Applicability~~

~~Applies to periodic testing requirements of the Auxiliary Feedwater System.~~ A.2

~~Objective~~

~~To verify the operability of the Auxiliary Feedwater System and its ability to respond properly when required.~~

~~Specification~~

IAW IST Program LA.2

1\. a. SR 3.7.5.2 Each auxiliary feedwater pump will be started manually from the control room at ~~monthly intervals on a staggered test basis (i.e., one pump per month, so that each pump is tested once during a 3 month period) with full flow established to the steam generators at least once per 24 months.~~ L.3

b. ~~The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.~~ LA.2 R.1

c. SEE ITS 3.7.7 Backup supply valves from the city water system will be tested at least once per 24 months. ~~[See Note A, below]~~

2\. SR 3.7.5.2 Acceptance levels of performance shall be that the pumps start, reach their required developed head ~~and operate for at least fifteen minutes.~~ A.7

3\. At least once per 24 months,

each automatic M.5

a. SR 3.7.5.3 Verify that the ~~recirculation~~ valve will actuate to its correct position.

b. SR 3.7.5.4 Verify that each auxiliary feedwater pump will start as designated automatically upon receipt of an auxiliary feedwater actuation test signal.

actuation A.8

Add SR 3.7.5.4, Note 1 L.2

Add SR 3.7.5.4, Note 2 M.1

~~Basis~~

~~The testing of the auxiliary feedwater pumps will verify their operability. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements.~~ A.1

~~Note A: Testing of the backup supply valves may be deferred until the next refueling outage (RO9), but no later than May 31, 1997.~~

Deleted by TSCR 98-043

Amendment No. ~~38~~, ~~125~~, ~~128~~, ~~172~~, 178

Superceded by Amendment 185
No impact on ITS 3.7.5
See Next Page

ITS 3.7.5 | R1

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater System.

Objective

To verify the operability of the Auxiliary Feedwater System and its ability to respond properly when required.

Specification

1. a. Each auxiliary feedwater pump will be started manually from the control room at monthly intervals on a staggered test basis (i.e., one pump per month, so that each pump is tested once during a 3 month period) with full flow established to the steam generators at least once per 24 months.

   b. The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

   c. Backup supply valves from the city water system will be tested at least once per 24 months.

2. Acceptance levels of performance shall be that the pumps start, reach their required developed head and operate for at least fifteen minutes.

3. At least once per 24 months,

   a. Verify that the recirculation valve will actuate to its correct position.

   b. Verify that each auxiliary feedwater pump will start as designated automatically upon receipt of an auxiliary feedwater actuation test signal.

Basis

The testing of the auxiliary feedwater pumps will verify their operability. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.5:
"Auxiliary Feedwater (AFW) System"

PART 5:

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

Status of NUREG 1431 Generic Changes for ITS 3.7.5

This ITS Specification is based on NUREG-1431 Specification No. 3.7.5
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
WOG-029	**101 R0**	CHANGE AFW PUMP TESTING FREQUENCY TO BE "IN ACCORDANCE WITH THE INSERVICE TESTING PROGRAM" NRC APPROVES	APPROVED/INCORPORATED	Incorporated	**T.1**
WOG-030	**029 RO**	REMOVE MODE 4 WHEN S/GS ARE RELIED UPON FROM THE MODES OF APPLICABILITY NRC REJECTS: TSTF ACCEPTS	Rejected by NRC	Not Incorporated	**N/A**
WOG-096	**268 R0**	REVISE THE FREQUENCY OF SR 3.7.5.5, AFW FLOW PATH VERIFICATION	APPROVED/NOT INCORP	Not Incorporated	**N/A**
WOG-112	**245 R1**	AFW TRAIN OPERABLE WHEN IN SERVICE	TSTF Review	Not Incorporated	**N/A**

3.7 PLANT SYSTEMS

<CTS> 3.7.5 Auxiliary Feedwater (AFW) System

<3.4.A.2> <DOC M.1> <DOC A.3>

LCO 3.7.5 [Three] AFW trains shall be OPERABLE.

[
-----------------------------NOTE-----------------------------
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

]

Insert: 3.7-11-01

<3.4.A> <3.4.E> <DOC M.1> <DOC A.4>

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC L.1> [A. One steam supply to turbine driven AFW pump inoperable.] <DOC M.2>	A.1 Restore steam supply to OPERABLE status.	[7 days **AND** 10 days from discovery of failure to meet the LCO]
<3.4.C.1> B. One AFW train inoperable in MODE 1, 2 or 3 [for reasons other than Condition A]. <DOC M.2>	B.1 Restore AFW train to OPERABLE status.	72 hours [**AND** 10 days from discovery of failure to meet the LCO]

(continued)

Typical

NUREG-1431 Markup Inserts
ITS SECTION 3.7.5 - Auxiliary Feedwater (AFW) System

INSERT: 3.7-11-01

capable of supporting the credited steam generator(s) | R.1

ACTIONS (continued)

<CTS>	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.C.1> <DOC M.3>	C. Required Action and associated Completion Time for Condition A [or B] not met.	C.1 Be in MODE 3.	6 hours
		AND	
		C.2 Be in MODE 4.	[18] hours
<3.4.C.2> <DOC M.3>	OR Two AFW trains inoperable in MODE 1, 2, or 3.		
<3.4.C.3> <DOC A.5> <3.4.E.2> <3.4.E.2a,b> <DOC A.5>	D. [Three] AFW trains inoperable in MODE 1, 2, or 3.	D.1 ----------NOTE---------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. --------------------------------- Initiate action to restore one AFW train to OPERABLE status.	Immediately
<DOC M.1>	E. Required AFW train inoperable in MODE 4.	E.1 Initiate action to restore AFW train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

<CTS>	SURVEILLANCE		FREQUENCY
<3.4.E> <DOC M.4>	SR 3.7.5.1	Insert: 3.7-13-01 Verify each AFW manual, power operated, and automatic valve in each water flow path, ~~[~~and in both steam supply flow paths to the steam turbine driven pump,~~]~~ that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
<DOC L.2> <DOC LA.2> <4.8.1.a> <4.8.2> <DOC A.7>	SR 3.7.5.2	-------------------NOTE------------------- Not required to be performed for the turbine driven AFW pump until ~~[~~24 hours~~]~~ after ≥ ~~[1000]~~ 600 psig in the steam generator. -- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	T.1 In accordance with Inservice Testing Program ~~[31] days on a STAGGERED TEST BASIS~~
<DOC M.1> <4.8.3.a> <DOC M.5> <4.8.1.b>	SR 3.7.5.3	-------------------NOTE------------------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. -- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[18]~~ 24 months

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.5 - Auxiliary Feedwater (AFW) System

INSERT: B 3.7-13-01

-------------------NOTE--------------------
Not applicable in MODE 4 when steam generator is relied upon for heat removal.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.5.4	-------------------NOTES------------------ 1. Not required to be performed for the turbine driven AFW pump until [24 hours] after ≥ [1000] psig in the steam generator. {600} 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. -- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	~~[18]~~ months 24
SR 3.7.5.5	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days

<DOC L.2>
<DOC M.1>
<4.8.3.b>
<DOC A.8>

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

via a Connection

at a point

(High Pressure Steam Dump)

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take suction ~~through separate and independent suction lines~~ from the condensate storage tank (CST) (LCO 3.7.6) and pump to the steam generator secondary side via ~~separate and independent connections~~ to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the steam bypass valves and recirculated to the CST. R.1

Insert B 3.7-23-01

, depending on steam conditions,

Capacity approaches

The AFW System consists of [two] motor driven AFW pumps and one steam turbine driven pump configured into [three] trains. Each motor driven pump provides [100]% of AFW flow capacity, and the turbine driven pump ~~provides~~ [200]% of the required capacity to the steam generators, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds [two] steam generators, ~~although each pump has the capability to be realigned from the control room to feed other steam generators~~. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

Insert: B 3.7-23-02

Any of the three pumps

The turbine driven AFW pump supplies a common header capable of feeding all steam generators ~~with DC powered control valves actuated to the appropriate steam generator by the Engineered Safety Feature Actuation System (ESFAS)~~. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry

~~(continued)~~

NUREG-1431 Markup Inserts
ITS SECTION 3.7.5 - Auxiliary Feedwater (AFW) System

INSERT: B 3.7-23-01

FSAR Section 10.2 (Ref. 1) describes this configuration as two pumping loops using two different types of motive power to the pumps. One auxiliary feedwater loop utilizes a steam turbine driven pump and the other utilizes two motor driven pumps. Technical specifications describe this configuration as three trains because DB.1

INSERT: B 3.7-23-02

Each of the steam generators can also be supplied by one of the two motor driven AFW pumps.

BASES

BACKGROUND (continued)

conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

Insert: B 3.7-24-01

~~The AFW System actuates automatically on steam generator water level—low-low by the ESFAS (LCO 3.3.2). The system also actuates on loss of offsite power, safety injection, and trip of all MFW pumps.~~

The AFW System is discussed in the FSAR, Section ~~10.4.9~~ 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus ~~3%~~ accumulation.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW System flow must also be available to account for flow losses such as pump recirculation and line breaks.

The limiting ~~Design Basis Accidents (DBAs) and transients for~~ events that require the AFW System are as follows:

Insert: B 3.7-24-02

~~a. Feedwater Line Break (FWLB); and~~

~~b. Loss of MFW.~~

~~In addition, the minimum available AFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident (LOCA).~~

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.5 - Auxiliary Feedwater (AFW) System

INSERT: B 3.7-24-01

The motor driven pumps are actuated by any one of the following:

1) Low-low level in any steam generator;
2) Loss of voltage (Non SI blackout) on 480 VAC bus 2A/3A (starts AFW Pump 31) and loss of voltage (Non SI blackout) on 480 VAC bus 6A (starts AFW Pump 33);
3) Safety Injection signal;
4) Auto trip of either main boiler feed pump;
5) Manual actuation from the Control Room; and
6) Manual actuation locally at the pump room.

The steam turbine driven pump is actuated by any one of the following:

1) Low-low level in two of the four steam generators;
2) Loss of voltage (Non SI blackout) on 480 VAC busses 2A/3A or 6A;
3) Manual actuation from the Control Room; and
4) Manual actuation locally at the pump room.

The steam driven AFW pump must be throttled manually in order to bring the unit up to speed after a start signal. In addition, the steam driven pump discharge flow control valves must be manually opened as necessary to provide adequate auxiliary feedwater flow.

INSERT: B 3.7-24-~~01~~ 02

a. small break loss of coolant accident;

b. loss of AC sources; and

c. loss of feedwater.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. In such a case, the ESFAS logic may not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly. One motor driven AFW pump would deliver to the broken MFW header at the pump runout flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.~~

actuates automatically

is available

~~The ESFAS automatically actuates~~ the AFW turbine driven pump ~~and associated power operated valves and controls~~ when required to ensure an adequate feedwater supply to the steam generators during loss of power. ~~DC~~ power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of ~~accidents~~ events that could result in overpressurization of the reactor coolant pressure boundary. ~~[~~Three~~]~~ independent AFW pumps ~~in [three] diverse trains~~ are required to be OPERABLE to ~~ensure the availability of RHR capability for all events accompanied by~~ ensure the capability to maintain the plant in hot shutdown with a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by ~~a different means,~~ a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

The AFW System is configured into ~~[~~three~~]~~ trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE ~~in [two] diverse paths~~, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with ~~redundant~~ two steam supplies from each of ~~[~~two~~]~~ main steam lines upstream

(continued)

BASES

LCO (continued)

of the MSIVs, and shall be capable of supplying AFW to ~~any~~ all of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

Insert: B3.7-26-01

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. ~~This is~~ because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory~~, lost as the unit cools to~~ MODE 4 conditions.

needed to achieve and maintain

a motor driven AFW pump may be needed to support

In MODE 4, ~~the AFW System may be used for~~ heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. The redundant OPERABLE steam supply to the turbine driven AFW pump;

b. The availability of redundant OPERABLE motor driven AFW pumps; and

c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.5 - Auxiliary Feedwater (AFW) System

INSERT: B 3.7-26-01

The motor driven AFW pump required to be OPERABLE in Mode 4 must be capable of supporting the SG(s) being credited as the redundant decay heat removal path in accordance with LCO 3.4.6, RCS Loops - MODE 4. This requirement ensures the ability to maintain the required level in the SG(s) (and decay heat removal capacity) during extended periods in Mode 4 with or without offsite power. Requiring only one OPERABLE AFW pump is acceptable | R.1 | R.1

BASES

ACTIONS

A.1 (continued)

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 [for reasons other than Condition A], action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

When Required Action A.1 [or B.1] cannot be completed within the required Completion Time, or if two AFW trains are

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within [18] hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

If all [three] AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops—MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

Insert: B3.7-29-01

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrigufal pump performance required by Section XI of the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) (only required at 3 month intervals) satisfies this requirement. ~~The [31] day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.~~ T.1

This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. When SG pressure is < 600 psig.

(continued)

INSERT: B 3.7-29-01

This SR is modified by a Note that states the SR is not required in MODE 4. Not performing this SR in MODE 4 is acceptable for the following reasons: AFW pumps are typically operated intermittently to keep the SGs filled when in MODE 4, the decay heat load is low; an RHR loop is required to be OPERABLE as the primary method of decay heat removal in Mode 4; and, the SG is required to be maintained at a level that ensures a significant inventory is available as a heat sink before the AFW pump is required to refill the SG. These factors ensure that a significant amount of time would be available to complete any valve realignments needed to refill a SG when in Mode 4.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The [18] month Frequency is acceptable based on operating experience and the design reliability of the equipment. 24

operated as necessary to maintain SG water level.

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is ~~already aligned and operating~~.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is ~~already operating~~ and the autostart function is not required. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. 24

operated as necessary

allows the test to be performed at rated conditions

This SR is modified by ~~[a]~~ [two] Note[s]. [Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral ~~is required because there is insufficient steam pressure to perform the test.~~] ~~[The]~~ Note [2] states that the SR is not required in MODE 4. [In MODE 4, the required pump is ~~already operating~~ and the autostart function is not required.] [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.]

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

[Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.]

[SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. (This SR is not required by those units that use AFW for normal startup and shutdown.)]

REFERENCES

1. FSAR, Section ~~10.4.9~~. 10.2
2. ASME, Boiler and Pressure Vessel Code, Section XI.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.6:
"Condensate Storage Tank (CST)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. CST inoperable.	A.1	Verify by administrative means OPERABILITY of City Water.	Immediately AND Once per 12 hours thereafter
	AND		
	A.2	Restore CST to OPERABLE.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4, without reliance on steam generator for heat removal.	18 hours

RAI-02

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST level is ≥ 360,000 gal.	12 hours

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves. The AFW steam driven pump operates with a continuous recirculation to the CST. The motor driven AFW pumps have recirculation controllers that recirculate flow to the CST, as necessary, to maintain a minimum required AFW pump flow.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam bypass (High Pressure Steam Dump) valves. The condensed steam is returned to the CST by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena. The CST is designed to Seismic Class I to ensure availability of the auxiliary feedwater supply. Auxiliary feedwater is also available from city water.

The condensate makeup system connects the 600,000 gallon capacity condensate storage tank to the main condenser. The condensate makeup system automatically supplies makeup water from the CST to the condenser if there is a low level in the condenser hotwell. Redundant, Category I, isolating valves will close the condenser makeup when the condensate storage tank level decreases to 360,000 gallons to reserve the required volume of condensate available to the auxiliary feedwater pumps sufficient to hold the plant at hot shutdown for 24 hours following a trip at full power.

(continued)

BASES

BACKGROUND (continued)

To ensure CST pressure is maintained within its design limits while limiting the amount of air in contact with the condensate, two Category I, 100% capacity breather valves are installed on the dome of the CST. CST venting is required for the CST to perform both its normal and emergency function. The venting function can be met by either of the CST breather valves or equivalent venting capacity.

A description of the CST is found in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and the minimum amount of water in the condensate storage tank is the amount needed to maintain the plant for 24 hours at hot shutdown following a trip from full power. When the condensate storage tank supply is exhausted, city water will be used.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36.

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat while in MODE 3 for 24 hours following a reactor trip from 102% RTP. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine. When the condensate storage tank supply is exhausted, city water will be used.

The CST level required is equivalent to a total volume of ≥ 360,000 gallons, which is based on holding the unit in MODE 3 for 24 hours. This basis is established in Reference 1. The CST total volume includes allowances for instrument accuracy and the unuseable volume in the CST.

(continued)

BASES

LCO (continued)	The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level. CST venting and pressure relief capability are required for the CST to perform both its normal and emergency function. The venting and pressure relief functions are satisfied by either of the CST breather valves or equivalent venting capacity.
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE. In MODE 5 or 6, the CST is not required because the AFW System is not required.

RAI-01

ACTIONS

A.1 and A.2

If the CST is not OPERABLE, the OPERABILITY of the backup supply (city water) should be verified by administrative means immediately and once every 12 hours thereafter. OPERABILITY of the backup auxiliary feedwater supply means that LCO 3.7.7, City Water, is met and includes verification that the flow paths from city water to the AFW pumps are OPERABLE. The CST must be restored to OPERABLE status within 7 days. The immediate Completion Time for verification of the OPERABILITY of the backup water supply ensures that Condition B is entered immediately if both the CST and City Water are inoperable. The 7 day Completion Time for restoration of the CST is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.

NYPA

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

If Condition B is entered when both the CST and City Water are not Operable, Conditions and Required Actions for LCO 3.7.5, Auxiliary Feedwater System, may be appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. FSAR, Section 10.2.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.6:
"CONDENSATE STORAGE TANK"

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
3.4-1	92	92	
3.4-2	92	92	
3.4-3	151	151	
3.4-4	151;1-18-95	151;1-18-95	

3.4 STEAM AND POWER CONVERSION SYSTEM

A.2

~~Applicability~~

~~Applies to the operating status of the Steam and Power Conversion System.~~

~~Objective~~

~~To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System operation is necessary to ensure the capability to remove decay heat from the core.~~

Specification Mode 1, 2 and 3 and Mode 4 when SG relied upon M.1

LCO 3.7.6 Applicability A. The reactor shall not be heated above 350°F unless the following conditions are met:

SEE ITS 3.7.1

(1) A minimum ASME Code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing). With up to three of the five main steam line safety valves per steam generator inoperable, heat-up above 350°F and power operation is permissible provided:

a) Within four hours,

the inoperable valve(s) is restored to operable status.

or

the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.4-1.

b) Otherwise the reactor shall be in hot shutdown within the next six hours and in cold shutdown within the following 30 hours.

SEE ITS 3.7.5 (2) Three out of three auxiliary feedwater pumps must be operable.

LCO 3.7.6

SR 3.7.6.1 (3) A minimum of 360,000 gallons of water in the condensate storage tank. Verify every 12 hours M.3

(4) ~~System piping and valves directly associated with the above components operable.~~ A.4

SEE ITS 3.7.2 (5) The main steam stop valves are operable and capable of closing in five seconds or less.

SEE ITS 3.7.5 (6) Two steam generators capable of performing their heat transfer function.

Add Required Action A.1 — M.2

Add Required Action B.2 — M.1

SEE ITS 3.7.7

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

LCO 3.7.6 ~~B.~~

Req. Act A.2

Req Act B.1, B.2

~~Except as modified by E below,~~ if during power operations any of the conditions ~~of 3.4-A~~ above, ~~except Items (1) and (2)~~ cannot be met within ~~48 hours~~ 7 days, the ~~operator shall start to shutdown and cool the reactor below 350°F using normal operation procedures.~~ Mode 3 in 6 hrs Mode 4 in 18 hrs — L.1 — A.3

SEE ITS 3.7.5

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

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

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

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

SEE RELOCTED CTS

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

SEE ITS 3.7.5

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

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

2) Within one hour either:

a) reopen the closed valve(s),

or

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

and

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

SEE ITS 3.7.5

If the above action cannot be taken, then:

a) maintain the plant in a safe stable mode which minimizes the potential for a reactor trip,

and

b) continue efforts to restore water supply to the auxiliary feedwater system,

and

c) notify the NRC within 24 hours regarding planned corrective action.

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system.

The twenty main steam safety valves have a total combined rated capability of 15,108,000 lbs/hr. The total full power steam flow is 12,974,500 lbs/hr.; therefore twenty (20) main steam safety valves will be able to relieve the total steam flow if necessary. The total relieving capacity of the twenty main steam line safety valves is 116% of the total secondary steam flow at 100% rated power (3025 Mwt). The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The operability of the twenty main steam line safety valves ensure that the secondary system pressure will be limited to within 110% of the design pressure of 1085 psig during the most severe anticipated system operational transient. A.1

Startup and/or power operation with inoperable main steam line safety valves is allowable within the limitation of Table 3.4-1. Operation with up to three of the five main steam line safety valves per steam generator inoperable is permissible if the maximum allowed power level is below the heat removing capability of the operable MSSVs. This is accomplished by restricting the reactor power level such that the heat input from the primary side will not exceed the heat removing capability of the operable MSSVs of the most limiting steam generator. The reduction in reactor power level is achieved by reducing the power range neutron flux high setpoint. The reactor trip setpoint reductions are derived on the following basis:

$$Hi\phi = (100 / Q) [(w_s h_{fg} N) / K]$$

A.1

Where:

- Hiϕ - Safety Analysis power range high neutron flux setpoint, percent.
- Q - Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt (3037 Mwt).
- K - Conversion factor, 947.82 $\frac{(Btu/sec)}{Mwt}$
- w_s - Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure, including tolerance and accumulation, as appropriate, in lb/sec. (w_s = 150 + 228.61 * (4 - V) lb/sec, where V = Number of inoperable safety valves in the steam line of the most limiting steam generator).
- h_{fg} - Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm (608.5 Btu/lbm).
- N - Number of loops in plant (4).

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tank is the amount needed for 24 hours at hot shutdown. When the condensate storage supply is exhausted, city water will be used.

The system piping and valves that are governed by Specification 3.4.A.(4) include the two (2) QA Category I, 100% cpacity breather valves installed on the dome of the Condensate Storage Tank (CST). The purpose of these valves is to ensure the CST pressure is within its design limits by providing both pressure relieving and vacuum break capability. Per Specification 3.4.B, if one (1) breather valve is inoperable, it must be returned to operability within 48 hours or the reactor must be shutdown and cooled to below 350°F using normal operating procedures. L.2 R.1

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove core decay heat after a reactor shutdown.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.6:
"Condensate Storage Tank (CST)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.7.6 - Condensate Storage Tank (CST)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.4.B requires the operator to shut down and cool the reactor below 350°F (i.e., Mode 4) using normal operation procedures if requirements for Operability of the Condensate Storage Tank cannot be met within the specified Completion Time. Under the same conditions, ITS 3.7.6, Required Action B.1 and B.2, require that the plant be in Mode 3 in 6 hours and Mode 4 in 18 hours (see 3.7.6, DOC M.1 for expanded

Applicability that includes reliance on SG for heat removal). The ITS Completion Time to reach Mode 3 within 6 hours is consistent with other ITS Completion Times and is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. The Completion Time of 18 hours to reach Mode 4 is not consistent with other ITS Completion Times in that additional time is provided in recognition that this cooldown may be conducted when the Auxiliary Feedwater function is degraded by the unavailability of the condensate storage tank and/or city water. The addition of specific Completion Times for shutdown and cooldown when CST or CW are not Operable is an administrative change with no impact on safety because the ITS Completion Times are consistent with a reasonable interpretation of the existing CTS 3.4.B requirements.

A.4 CTS 3.4.A.4 establishes requirements for Operability of system piping and valves directly associated with the Condensate Storage Tank. Valve lineups and verification of the Operability of active components associated with the CST and the flow path to the AFW pump suctions are verified as part of ITS LCO 3.7.5, Auxiliary Feedwater System. CTS 3.4.A.4 is deleted because it is a generic statement that does not provide any information or requirements specific to the CST. This is an administrative change with no impact on safety.

MORE RESTRICTIVE

M.1 CTS 3.4.A.3 requires that the Condensate Storage Tank (CST) is Operable as the primary water supply for the Auxiliary Feedwater System (AFW). City Water is the backup source of water to AFW system. CTS 3.4.A establishes the Applicability for CST Operability as whenever the reactor is heated above 350°F (i.e., Modes 1, 2 and 3). ITS LCO 3.7.6 maintains the requirement for Operability of the CST with an Applicability of Modes 1, 2 and 3; however, ITS LCO 3.7.6 expands the Applicability to include Mode 4 when a steam generator is relied upon for heat removal.

This change is needed because CST and CW Operability are both required

to support Auxiliary Feedwater System Operability; and, Applicability of ITS 3.7.5, Auxiliary Feedwater System, is expanded to include Mode 4 when a steam generator is relied upon for heat removal (see 3.7.5, DOC M.1). Therefore, the Applicability of ITS 3.7.6, Condensate Storage Tank, and ITS 3.7.7, City Water, must be expanded to include Mode 4 when a steam generator is relied upon for heat removal. The reasons and justification for the expanded Applicability of ITS 3.7.5 are addressed in ITS 3.7.5, Auxiliary Feedwater System, (see 3.7.5, DOC M.1).

In conjunction with this change, CTS 3.4.B requirements to shutdown and cool the reactor below 350°F (i.e., Mode 4) using normal operation procedures if requirements for Operability of City Water cannot be met is expanded in ITS 3.7.6, Required Action B.2, to require that the plant be placed in Mode 4, without reliance on steam generator for heat removal.

These changes are acceptable because operation of AFW in Mode 4 to feed a SG is consistent with operation of the AFW system as described in the FSAR. Therefore, operation and/or Operability of the supporting water supplies in the CST and CW are also consistent with operation of the AFW system as described in the FSAR. This more restrictive change does not introduce any operation which is un-analyzed while requiring added assurance that decay heat removal capability in Mode 4 is maintained. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.4.A.3 and CTS 3.4.A.7 require that the Condensate Storage Tank and City Water are Operable in Modes 1, 2 and 3 (see 3.7.7, DOC M.1). When either the CST or CW or both are not Operable, CTS 3.4.B allows 48 hours (see 3.7.7, DOC L.1) to restore both water supplies to Operable before a plant shutdown is required (see exception identified and explained below). Under the same conditions, ITS 3.7.6, Condensate Storage Tank, and ITS, 3.7.7, City Water, will not allow both the CST and CW to be inoperable at the same time. ITS 3.7.6, Required Action A.1, and ITS, 3.7.7, Required Action A.1, prevent and enforce this prohibition of simultaneous inoperability of both CST and CW by requiring immediate verification of the Operability of the alternate source (either CST or CW) and once per 12 hours thereafter if either the CST or CW are not

Operable. This change is needed to ensure AFW Operability by ensuring that either CST or CW is capable of supporting the AFW system. This more restrictive change does not introduce any operation which is un-analyzed while requiring added assurance that decay heat removal capability is maintained. Therefore, this change has no adverse impact on safety.

CTS 3.4.E provides Actions for a specific situation where neither the CST nor CW is (or can be) aligned to the AFW suction header. Although this condition appears to be simultaneous inoperability of both CST and CW, CTS 3.4.E appropriately specifies actions for 3 inoperable AFW pumps. Under the same conditions, ITS 3.7.5 surveillance requirements for AFW valve lineups would not be met for all three AFW pumps; therefore, ITS would also require the actions for 3 inoperable AFW pumps. Under the same conditions, surveillance requirements for ITS 3.7.6 and ITS 3.7.7 could, in many instances, be met and would not require that water sources be declared inoperable. Therefore, CTS 3.4.E is addressed with ITS 3.7.5, Auxiliary Feedwater System.

M.3 Neither CTS 3.4 nor CTS 4.8 establish any requirements for the verification of the Operability of the Condensate Storage Tank (CST) other than an implied requirement in CTS 3.4.A.3 to periodically verify the CST volume. ITS SR 3.7.6.1 is added to require verification every 12 hours that the CST contains a reserve of condensate for the auxiliary feedwater pumps sufficient to hold the plant at hot shutdown for 24 hours following a trip at full power. This change is acceptable because CST level verification is a passive indication of CST availability that is not currently required. This more restrictive change does not introduce any operation which is un-analyzed while requiring added assurance that decay heat removal capability is maintained. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.4.A.3 and CTS 3.4.A.7 require that the Condensate Storage Tank and City Water are Operable in Modes 1, 2 and 3 (see 3.7.7, DOC M.1). When

either the CST or CW or both are not Operable, CTS 3.4.B allows 48 hours to restore both water supplies to Operable before a plant shutdown is required. Under the same conditions, ITS 3.7.6, Condensate Storage Tank, and ITS 3.7.7, City Water, will not allow both the CST and CW to be inoperable at the same time (see 3.3.7, DOC M.2): however, ITS 3.7.6, Required Action A.2, and ITS 3.7.7, Required Action A.2, are, in part, less restrictive because the ITS extends the time that either the CST or CW (but not both) can be inoperable from 48 hours to 7 days.

Extending the allowable out of service time for an inoperable CST or CW supply (but not both) from 48 hours to 7 days is acceptable because either source of water is capable of meeting the minimum assumptions of the accident analysis although the CST with a safety grade source of water is the preferred source for feeding the SGs. The 7 day Completion Time for restoration of both the CST and CW recognizes that the CST is the preferred source of water to the SGs and should be restored promptly, the desirability of maintaining city water as a backup source to the CST, and the low probability of an event occurring during this time period requiring the AFW and the associated water supply. This change has no significant adverse impact on safety because, in conjunction with this change, ITS eliminates the CTS 3.4.B Actions that allow CST and CW to be inoperable simultaneously for up to 48 hours.

L.2 The CTS Bases for CTS 3.4.A.(4) includes a requirement that is not part of CTS 3.4.A.(4). Specifically, the Bases require that the CST piping and valves that are governed by Specification include the two (2) QA Category I, 100% capacity breather valves installed on the dome of the Condensate Storage Tank (CST). The purpose of these valves is to ensure the CST pressure is within its design limits by providing both pressure relieving and vacuum break capability. Per Specification 3.4.B, if one (1) breather valve is inoperable, it must be returned to operability within 48 hours or the reactor must be shutdown and cooled to below 350oF using normal operating procedures.

ITS 3.7.6 does not include a specific requirement for the breather valves installed on the dome of the CST: however, the Bases specify that CST venting and pressure relief capability are required for the CST to

perform both its normal and emergency function. The venting and pressure relief functions are satisfied by either of the CST breather valves or equivalent venting capacity. Use of an open vent instead of the breather valves is acceptable because an open vent provides equivalent pressure relieving and vacuum break capability and the role of the breather valves in limiting oxygen contact with the condensate is not a safety function. Eliminating the requirement for reactor shutdown within 48 hours is acceptable because one breather valve is capable of performing the required safety function and the breather valves are reliable. Additionally, the IP3 corrective action program will ensure that an inoperable breather valve is restored to service if it is determined to be inoperable. Therefore, this change has no significant adverse impact on safety.

REMOVED DETAIL

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.6:
"Condensate Storage Tank (CST)"

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 3.7.6 - Condensate Storage Tank (CST)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards considerations are discussed below.

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

   This change extends the time that either the Condensate Storage Tank (CST) or City Water (CW) (but not both) can be inoperable from 48 hours to 7 days. The Condensate Storage Tank (CST) and City Water (CW) are the primary and backup source of water for the Auxiliary Feedwater System. This change will not result in a significant increase in the probability of an accident previously evaluated because the status of neither the CST nor CW has any effects on the initiators of any accident previously evaluated. This change will not result in a significant increase in the consequences of an accident previously evaluated because the CST is maintained with a volume of water sufficient to hold the plant at hot shutdown for a minimum of 24 hours and CW is capable of providing sufficient water decay heat removal indefinitely. If an accident occurs when CW is not available and cannot be restored within 24 hours, the 24 hour supply of water in the CST provides sufficient time to either complete a plant cooldown (and establish RHR as the primary decay heat removal mechanism) or establish an alternate supply of water to the AFW suction. This change has no effect on safety because, in conjunction with this change, ITS eliminates the CTS 3.4.B Actions that allow CST and CW to be inoperable simultaneously for up to 48 hours.

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

   The proposed change will not involve any physical changes to systems,

structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the CST is maintained with a volume of water sufficient to hold the plant at hot shutdown for a minimum of 24 hours and CW is capable of providing sufficient water indefinitely. If an accident occurs when CW is not available and cannot be restored within 24 hours, the 24 hour supply of water in the CST provides sufficient time to either complete a plant cooldown (and establish RHR as the primary decay heat removal mechanism) or establish an alternate supply of water to the AFW suction. This change has no significant adverse impact on safety because, in conjunction with this change, ITS eliminates the CTS 3.4.B Actions that allow CST and CW to be inoperable simultaneously for up to 48 hours.

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards considerations are discussed below.

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

   ITS 3.7.6 does not include a specific requirement for the breather valves installed on the dome of the CST; however, the Bases specify that CST venting and pressure relief capability are required for the CST to perform both its normal and emergency function. The venting and

pressure relief functions are satisfied by either of the CST breather valves or equivalent venting capacity. This change will not result in a significant increase in the probability of an accident previously evaluated because the status of neither the CST nor CW has any effects on the initiators of any accident previously evaluated. This change will not result in a significant increase in the consequences of an accident previously evaluated because use of an open vent instead of the breather valves is acceptable because an open vent provides equivalent pressure relieving and vacuum break capability and the role of the breather valves in limiting oxygen contact with the condensate is not a safety function. Eliminating the requirement for reactor shutdown within 48 hours is acceptable because one breather valve is capable of performing the required safety function and the breather valves are reliable. Additionally, the IP3 corrective action program will ensure that an inoperable breather valve is restored to service if it is determined to be inoperable.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because use of an open vent instead of the breather valves is acceptable because an open vent provides equivalent pressure relieving and vacuum break capability and the role of the breather valves in limiting oxygen contact with the condensate is not a safety function. Eliminating the requirement for reactor shutdown within 48 hours is acceptable because one breather valve is capable of performing the required safety function and the breather valves are reliable. Additionally, the IP3 corrective action program will ensure that an inoperable breather valve is restored to service if it is determined to be inoperable.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.6:
"Condensate Storage Tank (CST)"

PART 5:

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

Status of NUREG 1431 Generic Changes for ITS 3.7.6

This ITS Specification is based on NUREG-1431 Specification No. 3.7.6 as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
CEOG-052	140 R1	CORRECT CONDENSATE STORAGE TANK LCO AND CRITERIA	APPROVED/INCORPORATED	Incorporated	T.1
CEOG-079	174 R0	ADD BASES FOR LCO 3.7.6, ACTIONS A.1 AND A.2	APPROVED/INCORPORATED	Incorporated	T.2
WOG-030	029 RO -	REMOVE MODE 4 WHEN S/GS ARE RELIED UPON FROM THE MODES OF APPLICABILITY NRC REJECTS: TSTF ACCEPTS	Rejected by NRC	Not Incorporated	N/A

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST)

<3.4.A.3>
<3.4.A.4>
<Doc A.4>

LCO 3.7.6 The CST ~~level~~ shall be ~~≥ [110,000 gal]~~ OPERABLE. T.1

<3.4.A>
<Doc M.1>

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<Doc M.2> A. CST ~~level not within limit~~ inoperable.	A.1 Verify by administrative means OPERABILITY of ~~backup water supply~~ City Water R.1	~~4 hours~~ Immediately DB.2 T.1 AND Once per 12 hours thereafter
	AND	
<3.4.B> <Doc L.1>	A.2 Restore CST ~~level~~ to ~~within limit~~ OPERABLE	7 days T.1
<3.4.B> <Doc A.3> B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
<3.4.B> <Doc A.3> <Doc M.1>	B.2 Be in MODE 4, without reliance on steam generator for heat removal.	~~[~~18~~]~~ hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<3.4.A.3> <DOC M.3>	SR 3.7.6.1 Verify the CST level is ≥ ~~[110,000 gal]~~. {360,000 gal}	12 hours

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND

The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves. The AFW pumps operate with a continuous recirculation to the CST.

steam driven

Insert: B 3.7-32-01

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam bypass valves. The condensed steam is returned to the CST by the condensate ~~transfer~~ pump. This has the advantage of conserving condensate while minimizing releases to the environment.

(High Pressure Steam Dump)

Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena ~~including missiles that might be generated by natural phenomena~~. The CST is designed to Seismic ~~Category~~ I to ensure availability of the feedwater supply. Feedwater is also available from alternate sources.

city water

auxiliary

Class

Insert: B3.7-32-02

A description of the CST is found in the FSAR, Section ~~[9.2.6]~~ (Ref. 1).

10.2

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat ~~and to cool down the unit~~ following all events in the accident analysis ~~as discussed in the FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs, followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate.~~

Insert: B 3.7-32-03

~~The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite~~

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.6 - Condensate Storage Tank (CST)

INSERT: B 3.7-32-01

The motor driven AFW pumps have recirculation controllers that recirculate flow to the CST, as necessary, to maintain a minimum required AFW pump flow.

INSERT: B 3.7-32-02

The condensate makeup system connects the 600,000 gallon capacity condensate storage tank to the main condenser. The condensate makeup system automatically supplies makeup water from the CST to the condenser if there is a low level in the condenser hotwell. Redundant, Category I, isolating valves will close the condenser makeup when the condensate storage tank level decreases to 360,000 gallons to reserve the required volume of condensate available to the auxiliary feedwater pumps sufficient to hold the plant at hot shutdown for 24 hours following a trip at full power.

To ensure CST pressure is maintained within its design limits while limiting the amount of air in contact with the condensate, two Category I, 100% capacity breather valves are installed on the dome of the CST. CST venting is required for the CST to perform both its normal and emergency function. The venting function can be met by either of the CST breather valves or equivalent venting capacity.

INSERT: B 3.7-32-03

and the minimum amount of water in the condensate storage tank is the amount needed to maintain the plant for 24 hours at hot shutdown following a trip from full power. When the condensate storage tank supply is exhausted, city water will be used.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~power. Single failures that also affect this event include the following:~~

~~a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and~~

~~b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).~~

~~These are not usually the limiting failures in terms of consequences for these events.~~

~~A nonlimiting event considered in CST inventory determinations is a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, since the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.~~

Criteria 2 and 3

10 CFR 50.36

The CST satisfies Criterion 3 of ~~the NRC Policy Statement~~.

T.1

LCO

While in Mode 3

24 hours

Insert: B3.7-33-01

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for ~~[30 minutes]~~ following a reactor trip from 102% RTP. ~~and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure.~~ In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine. ~~or before isolating AFW to a broken line~~

360,000

Total

24

Insert: B3.7-33-02

The CST level required is equivalent to a ~~usable~~ volume of ≥ ~~[110,000~~ gallons], which is based on holding the unit in MODE 3 for ~~[2]~~ hours. ~~followed by a cooldown to RHR entry conditions at [75]°F/hour.~~ This basis is established in Reference ~~4~~ 1 ~~and exceeds the volume required by the accident analysis~~.

R.1

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.6 - Condensate Storage Tank (CST)

<u>INSERT: B 3.7-33-01</u>

When the condensate storage tank supply is exhausted, city water will be used.

<u>INSERT: B 3.7-33-02</u>

The CST total volume includes allowances for instrument accuracy and the unuseable volume in the CST.

Insert: B3.7-34-04 | R.1

BASES

LCO (continued) The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2 OPERABLE auxiliary T.1

If the CST level is not ~~within limits~~, the OPERABILITY of the backup supply should be verified by administrative means ~~within 4 hours~~ and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply ~~must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available~~. The CST must be restored to OPERABLE status within 7 days, ~~because the backup supply may be performing this function in addition to its normal functions~~. ~~The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply~~. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST.

(City water)

immediately

Insert: B3.7-34-01

Insert: B 3.7-34-02

for restoration of the CST

DB.1

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within ~~[~~18~~]~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

Insert: B 3.7-34-03

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.6 - Condensate Storage Tank (CST)

INSERT: B 3.7-34-01 (Rev 1)

means that LCO 3.7.7, City Water, is met and includes verification that the flow paths from city water to the AFW pumps are OPERABLE.

NYPA

INSERT: B 3.7-34-02

The immediate Completion Time for verification of the OPERABILITY of the backup water supply ensures that Condition B is entered immediately if both the CST and City Water are inoperable.

INSERT: B 3.7-34-03

If Condition B is entered when both the CST and city Water are not Operable, Conditions and Required Actions for LCO 3.7.5, Auxiliary Feedwater System, may be appropriate.

INSERT: B 3.7-34-04

CST venting and pressure relief capability are required for the CST to perform both its normal and emergency function. The venting and pressure relief functions are satisfied by either of the CST breather valves or equivalent venting capacity.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. ~~(The required CST volume may be single value or a function of RCS conditions.)~~ The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. FSAR, Section ~~[9.2.6]~~. 10.2
2. ~~FSAR, Chapter [6].~~
3. ~~FSAR, Chapter [15].~~

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.6:
"Condensate Storage Tank (CST)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.7.6 - Condensate Storage Tank (CST)

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. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DB.2 IP3 LCO 3.7.6, Required Action A.1 and the supporting Bases, differs from NUREG-1431, Rev 1, in that IP3 will require immediate (versus 4 hours allowed in NUREG 1431) verification of the Operability of the alternate water source if either the CST or CW are not Operable. This change is needed and is acceptable because IP3 has an LCO that governs Operability of the backup water source. Specifically, IP3 LCO 3.7.7, City Water, ensures the Operability of the backup water source whenever the CST is required to be Operable. Therefore, the immediate Completion Time for verification of the Operability of the backup water supply ensures that Condition B is entered immediately if neither the CST nor City Water is OPERABLE.

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.7.6 - Condensate Storage Tank (CST)

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

T.1 This change incorporates Generic Change TSTF-140 (CEOG-52) which revises LCO 3.7.6, "Condensate Storage Tank" from requiring a specific CST volume to requiring that the CST be operable. The 10 CFR 50.36.(c).(2).(ii) criteria are also corrected to be consistent with the LCO. This change is needed because LCO 3.7.6 requires that "The CST level shall be >= ([110,000) gal." This presentation is inconsistent with other ITS LCOs in that it does not address Operability. The LCO is revised to state, "The CST shall be OPERABLE." The details of what constitutes Operability are given in the Bases. The requirement to maintain and periodically verify CST level remains in SR 3.7.6.1 and continues to be an Operability requirement in accordance with SR 3.0.1. Action A is revised from "CST level not within limit" to "CST inoperable". This presentation is consistent with similar Specifications. The Applicable Safety Analysis section states that CST volume meets Criterion 3 (mitigation), when it also meets Criterion 2 (process variable assumed as an initial condition). This has also been corrected. These changes make the Specifications consistent with the ITS rules and presentation without making any change to the existing requirements.

T.2 This change incorporates Generic Change TSTF-174 (CEOG-79) which revises the Bases for 3.7.6 Actions A.1 and A.2 to describe the frequency for performing the backup water supply verification. This description was added to bring the Bases in compliance with the NUREG format.

DIFFERENCE FOR ANY REASON OTHER THAN ABOVE

None

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.8:
"Component Cooling Water (CCW) System"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.7.8 - Component Cooling Water (CCW) System

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.3.E.1 specifies the Applicability for Component Cooling System as whenever the reactor is above cold shutdown (i.e., Modes 1, 2, 3 and 4). ITS 3.7.8 maintains this Applicability by requiring that the Component Cooling Water (CCW) System be Operable in Modes 1, 2, 3 and 4. This is an administrative change with no impact on safety because there is no change to the existing CTS Applicability.

A.4 CTS 3.3.E includes requirements for component cooling water (CCW) pumps and heat exchangers and CTS 3.1.A.1.c includes requirements for the RHR decay heat removal capability in Mode 4. If an inoperable CCW loop caused an RHR loop to be inoperable when RHR is required for decay heat removal, CTS would require that both CCW and the affected RHR heat exchanger be declared inoperable. Under the same conditions (inoperability of CCW loop causes RHR inoperability), ITS LCO 3.0.6 specifies that only Required Actions for an inoperable CCW loop are required. Therefore, ITS 3.7.8, Required Action A.1, is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops – Mode 4," must be entered if an inoperable CCW loop results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. This is an administrate change with no impact on safety because both CTS and ITS are designed to ensure that appropriate actions are taken if an inoperable CCW loop causes RHR to be inoperable if RHR is being relied upon for decay heat removal. This is an administrative change with no impact on safety because there is no change to the existing requirements.

A.5 CTS 3.3.E.2.c specifies that a CCW heat exchanger or other passive component may be out of service for 48 hours if "the system will still operate at design accident capability." This statement is deleted because ITS 3.7.8 is designed to ensure that CCW is operated consistent with the assumptions of the FSAR and WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95°F at IP3," regarding minimum heat removal capability even when redundant capability is not Operable. Removal of this statement is needed because Technical Specifications are designed to ensure that operators are not required to make a determination that "the system will still operate at design accident capability." This is an administrative change with no impact on safety because operating within the restrictions established by LCO and associated Required Actions ensures that "the system will still operate at design accident capability."

A.6 CTS 3.3.E does not include any requirements or guidance regarding the

effect on CCW Operability of isolation of CCW flow to individual components. ITS SR 3.7.8.1, the requirement for monthly valve lineups, is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the Operability of the CCW System (i.e., only the affected component is inoperable). This is an administrative change with no impact on safety because it is a reasonable interpretation of the existing CTS requirements.

MORE RESTRICTIVE

M.1 CTS 3.3.E.3.a specifies that if requirements for CCW are not met when the reactor is critical and CCW is not restored to Operable within the specified restoration time, then a shutdown to the cold shutdown condition must be initiated immediately. However, CTS 3.3.E.3.b specifies that if requirements for CCW are not met when the reactor is subcritical and CCW is not restored to Operable within the specified restoration time, then plant shutdown to Mode 5 may be delayed an additional 48 hours as long as the reactor coolant system temperature and pressure are not increased more than 25°F and 100 psi, respectively, over existing values.

Under the same conditions, ITS LCO 3.7.8, Required Actions B.1 and B.2, require an immediate plant shutdown to Mode 5 regardless of the status of the reactor (critical or subcritical) when the inoperability with CCW is identified. This change is needed because CTS 3.3.E.3.a and CTS 3.3.E.3.b, are ambiguous and potentially contradictory. Specifically, prior to the completion of CTS 3.3.E.3.a, plant operators could exit CTS 3.3.E.3.a Actions by entering CTS 3.3.E.3.b Actions and have an additional 48 hour to complete the shutdown. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring a prompt shutdown when minimum CCW capacity and/or redundancy cannot be restored within the specified completion time. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.3.E includes requirements for CCW Operability; however, there are

no explicit requirements for periodic verification of the key aspects to CCW Operability. ITS SR 3.7.8.1, a requirement for valve lineups every 92 days; SR 3.7.8.2, a requirement to verify proper automatic operation of the CCW valves every 24 months; and, SR 3.7.8.3, a requirement to verify automatic operation of the CCW pumps every 24 months are added to ensure that CCW will respond as required during an accident. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification that pumps will start (if not already running) and periodic verification that flow paths exist for CCW operation. Therefore, this change has no adverse impact on safety.

LESS RESTRICTIVE

L.1 CTS 3.3.E.2, CTS 3.3.E.2.a and CTS 3.3.E.2.c allow either (but not both) one CCW pump or one CCW heat exchanger to be inoperable with an Allowable Out of Service Time (AOT) of 24 hours for a pump and 48 hours for a heat exchanger or other passive component. Although CCW is normally operated cross connected, ITS LCO 3.7.8 establishes requirements for two CCW loops with a loop consisting of one pump and one heat exchanger. In conjunction with this change, ITS LCO 3.7.8 will allow both a pump and/or heat exchanger in the same loop to be inoperable at the same time and will extend the AOT for a pump and/or heat exchanger to 72 hours. This change is acceptable because one CCW pump and one CCW heat exchanger are adequate to perform the post accident heat removal function in accordance with WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP-3". Additionally, since CCW pumps and heat exchangers are running during normal plant operation, a failure of the remaining CCW pump would be promptly identified and appropriate actions taken. Therefore, this change does not have a significant impact on safety.

L.2 CTS 3.3.E.3.a specifies that if requirements for CCW are not met and CCW is not restored to Operable within the specified restoration time, then the plant must be in hot shutdown (Mode 3) within 4 hours and in cold shutdown (Mode 5) within the following 24 hours (i.e., 28 hours after

discovery of failure to meet requirements). Under the same conditions, ITS LCO 3.7.8, Required Actions B.1 and B.2, allow 6 hours to reach Mode 3 and 36 hours to reach Mode 5. This change is acceptable because these times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Additionally, there is a low probability of a DBA occurring during the additional 8 hours allowed to reach Mode 5. Finally, the ITS Completion Times allowed to achieve cold shutdown (Mode 5) are more restrictive than the allowances provided by CTS 3.3.E.3.b which can be interpreted as adding 48 hours to the CTS 3.3.E.3.a requirements for reaching Mode 5 (see 3.7.8, DOC M.1). Therefore, this change has no impact on safety.

L.3 CTS 3.3.E.1.b requires that two auxiliary component cooling pumps, one per each recirculation pump, together with their associated piping and valves, are operable; and CTS 3.3.E.2.b specifies that two auxiliary component cooling pumps serving the same recirculation pump may be out of service, provided at least one is restored to an operable status within 24 hours and at least one auxiliary component cooling pump serving the other recirculation pump is operable.

ITS LCOs 3.5.2 and CTS 3.7.8 recognize that the auxiliary component cooling pumps covered by CTS 3.3.E.1.b and CTS 3.3.E.2.b are support systems for the Containment Recirculation pumps which are governed by ITS LCO 3.5.2. Therefore, the Bases for ITS LCO 3.5.2 specifies that Containment Recirculation pump OPERABILITY requires the functional availability of an associated auxiliary component cooling water pump and CTS 3.3.E.1.b and CTS 3.3.E.2.b are deleted. Therefore, if at least one auxiliary component cooling pump capable of supporting each Containment Recirculation is not Operable, the supported Containment Recirculation pump is not Operable and the requirements of ITS LCO 3.5.2 ensure the appropriate requirements are applies. Therefore, this change has no impact on safety.

DISCUSSION OF CHANGES
ITS SECTION 3.7.8 - Component Cooling Water (CCW) System

REMOVED DETAIL

LA.1 CTS 3.3.E.1.a and CTS 3.3.E.1.c define the requirements for Operability of the CCW System and require 2 (of 3) CCW pumps, 2 CCW heat exchangers, and associated piping and valves. ITS LCO 3.7.8 maintains this requirement by requiring the Operability of 2 CCW loops; however, the details about what constitutes a loop are relocated to ITS 3.7.8 Bases and system descriptions are found in the FSAR.

This change is acceptable because LCO 3.7.8 maintains the existing requirement for the Operability of two loops of CCW; therefore, there is no change to the existing requirements and no change to the level of safety of facility operation.

This change, which allows the description of the design of the CCW system to be maintained in the FSAR and the detailed description of the requirements for Operability of these systems to be maintained in the ITS Bases, is consistent with the approach used in NUREG-1431 for all Limiting Conditions for Operation (LCOs). This approach is acceptable because the requirements of 10 CFR 50.59, Changes, Tests and Experiments, and ITS 5.5.13, Technical Specifications (TS) Bases Control Program, are designed to assure that changes to the FSAR and ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement FSAR changes in accordance with 10 CFR 50.59 and ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of FSAR and Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because 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.

DISCUSSION OF CHANGES
ITS SECTION 3.7.8 - Component Cooling Water (CCW) System

LA.2 CTS Table 4.1-2, Item 7, requires monthly verification of CCW levels of gross activity, pH and corrosion inhibitor. ITS LCO 3.7.8 maintains the requirements for CCW Operability; however, requirements for periodic verification of CCW levels of gross activity, pH and corrosion inhibitor are relocated to the Technical Requirements Manual (TRM).

This change is acceptable because CCW chemistry requirements do not meet any of the 10 CFR 50.36 criteria for retention in the Technical Specifications. Specifically, CCW gross activity is monitored as an early and sensitive indicator of CCW system leakage and low levels of gross activity have no direct impact on CCW Operability. CCW pH and corrosion inhibitor levels are monitored because maintaining CCW water chemistry reduces long term degradation of system materials. CCW pH and corrosion inhibitor levels have no direct impact on CCW Operability.

The Quality Assurance Plan will be revised to specify that requirements in the TRM are part of the facility as described in the FSAR and that changes to the TRM can be made only in accordance with the requirements of 10 CFR 50.59. Therefore, this change is acceptable because there is no change to the existing requirements by the relocation of requirements to the TRM and future changes to the TRM will be controlled in accordance with 10 CFR 50.59.

This change is a less restrictive administrative change with no impact on safety because ITS 3.7.8 maintains the requirements to have CCW Operable and maintains the requirements to perform periodic verification that demonstrates CCW Operability. Therefore, maintaining CCW chemistry requirements in the TRM has no significant adverse impact on safety.

LA.3 CTS 4.5.B.1.a and CTS 4.5.B.1.b require starting the auxiliary component cooling water pumps and operating for at least 15 minutes at the required pressure every quarter. This requirement is relocated to the Inservice Test (IST) Program. This change is acceptable because ITS LCOs 3.5.2 and CTS 3.7.8 recognize that the auxiliary component cooling pumps covered by CTS 3.3.E.1.b and CTS 3.3.E.2.b are support systems for the Containment Recirculation pumps which are governed by ITS LCO 3.5.2. Therefore, the Bases for ITS LCO 3.5.2 specifies that Containment

Recirculation pump operability requires the functional availability of an associated auxiliary component cooling water pump and CTS 3.3.E.1.b and CTS 3.3.E.2.b are deleted.

Allowing the testing the auxiliary component cooling pumps to be governed by the Inservice Test (IST) Program is acceptable because there are redundant auxiliary component cooling pumps for each of the redundant Containment Recirculation pumps. Additionally, the IST Program is required by ITS 5.5.7 and provides controls for inservice testing of ASME Code Class 1, 2, and 3 components.

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. Therefore, maintaining the requirement that auxiliary component cooling pumps must be functional to support the Containment Recirculation pumps in ITS 3.5.2 and maintaining the requirement for periodic testing of pumps in the IST Program required by ITS 5.5.7 provides a high degree of assurance that these pumps will be tested and maintained to ensure ECCS 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.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.8:
"Component Cooling Water (CCW) System"

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 3.7.8 - Component Cooling Water (CCW) System

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change will allow both a Component Cooling Water (CCW) pump and/or heat exchanger to be inoperable at the same time and will extend the (AOT) for a pump and/or heat exchanger to 72 hours from the current 24 hours for a pump and 48 hours for a heat exchanger or other passive component. This change will not result in a significant increase in the probability of an accident previously evaluated because CCW status is not assumed as the initiator of any accident previously evaluated. This change will not result in a significant increase in the consequences of an accident previously evaluated because one CCW pump and one CCW heat exchanger are adequate to perform the post accident heat removal function in accordance with WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP-3". Additionally, since CCW pumps and heat exchangers are running during normal plant operation, a failure of the remaining CCW pump would be promptly identified and appropriate actions taken.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because one CCW pump and one CCW heat exchanger are adequate to perform the post accident heat removal function in accordance with WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increased to 95° at IP-3".

LESS RESTRICTIVE
("L.2" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change extends the time allowed to complete a plant shutdown when requirements for Component Cooling Water (CCW) are not met from 4 hours to 6 hours to reach Mode 3 and from 28 hours to 36 hours to reach Mode 5. This change will not result in a significant increase in the probability of an accident previously evaluated because the amount of time allowed to perform a controlled shutdown when requirements are not met is not assumed as the initiator of any accident previously evaluated. This change will not result in a significant increase in the consequences of an accident previously evaluated because extending the amount of time allowed to shutdown has no effect on any system used to mitigate an accident and the additional 8 hours is a relatively small amount of additional time to reach the shutdown condition.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because extending the amount of time allowed to shutdown has no effect on any system used to mitigate an accident and the additional 2 hours is a relatively small amount of additional time to reach the shutdown condition.

LESS RESTRICTIVE
("L.3" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change deletes the LCO and associated Required Actions for the auxiliary component cooling pumps covered by CTS 3.3.E.1.b and CTS 3.3.E.2.b. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated because ITS LCOs 3.5.2 and CTS 3.7.8 recognize that the auxiliary component cooling pumps covered by CTS 3.3.E.1.b and CTS 3.3.E.2.b are support

systems for the Containment Recirculation pumps which are governed by ITS LCO 3.5.2. Therefore, the Bases for ITS LCO 3.5.2 specify that Containment Recirculation pump OPERABILITY requires the functional availability of an associated auxiliary component cooling water pump and CTS 3.3.E.1.b and CTS 3.3.E.2.b are deleted. Therefore, if at least one auxiliary component cooling pump capable of supporting each Containment Recirculation is not Operable, the supported Containment Recirculation pump is not Operable and the requirements of ITS LCO 3.5.2 ensure the appropriate requirements are applied.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because ITS LCOs 3.5.2 and CTS 3.7.8 recognize that the auxiliary component cooling pumps covered by CTS 3.3.E.1.b and CTS 3.3.E.2.b are support systems for the Containment Recirculation pumps which are governed by ITS LCO 3.5.2. Therefore, the Bases for ITS LCO 3.5.2 specify that Containment Recirculation pump OPERABILITY requires the functional availability of an associated auxiliary component cooling water pump and CTS 3.3.E.1.b and CTS 3.3.E.2.b are deleted. Therefore, if at least one auxiliary component cooling pump capable of supporting each Containment Recirculation is not Operable, the supported Containment Recirculation pump is not Operable and the requirements of ITS LCO 3.5.2 ensure the appropriate requirements are applied.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.8:
"Component Cooling Water (CCW) System"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<Doc A.6> SR 3.7.7.1 8	------------------NOTE------------------ Isolation of CCW flow to individual components does not render the CCW System inoperable. --	
<Doc M.2>	~~Verify each CCW manual, power operated, and~~ automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	~~31~~ 92 days
<Doc M.2> SR 3.7.7.2 8	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18~~ 24 months
<Doc M.2> SR 3.7.7.3 8	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	~~[18]~~ 24 months

X.1

R.1

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.7.8 - Component Cooling Water (CCW) System

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. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCES FOR ANY REASON OTHER THAN ABOVE

X.1 IP3 CTS has no requirement to periodically verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related

equipment, that is not locked, sealed, or other wise secured in position, is in the correct position. IP3 is voluntarily adopting this STS SR 3.7.8.1, at a frequency of 92 days versus the 31 day frequency in NUREG-1431. The 31 day frequency would be a significant burden without commensurate contribution to plant safety. The 92 day frequency is consistent with the current frequency for pump tests that include verifying proper valve position and operating experience has shown the current frequency to be adequate. The CCW headers are normally cross connected during normal and emergency operation, while the cooling loads are divided between the two loops so that each loop is capable of supplying the necessary service to support normal operation heat loads or accident heat loads. Any non-essential required service water pump can be used to support either or both CCW heat exchangers. This frequency is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification of proper valve lineup at a frequency consistent with good practice.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.9:
"Service Water System (SWS)"

PART 1:

Indian Point 3
Improved Technical Specifications and Bases

3.7 PLANT SYSTEMS

3.7.9 Service Water System (SWS)

LCO 3.7.9 Three pumps and required flow path for the essential SWS header shall be Operable;

AND,

Two pumps and required flow path for the nonessential SWS header shall be Operable.

APPLICABILITY: MODES 1, 2, 3, and 4.

-----NOTE-----
If LCO 3.7.9 will be met after the essential and non-essential header are swapped, then LCO 3.0.3 is not applicable for 8 hours while swapping the essential SWS header with the nonessential SWS header.

RAI-01

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required SWS pump on essential header inoperable.	A.1	Establish 3 OPERABLE SWS pumps on the essential SWS header.	72 hours
B. One required SWS pump on nonessential header inoperable.	B.1	Establish 2 OPERABLE SWS pumps on the nonessential SWS header.	72 hours

RAI-01

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One EDG ESFAS Service Water valve inoperable.	C.1 Restore both EDG ESFAS Service Water valves to OPERABLE status.	12 hours
D. One FCU ESFAS Service Water valve inoperable.	D.1 Restore both FCU ESFAS Service Water valves to OPERABLE status.	12 hours
E. Required Action and associated Completion Time of Condition A, B, C or D not met.	E.1 Be in MODE 3	6 hours
	<u>AND</u>	
	E.2 Be in MODE 5.	36 hours

RAI-01

RAI-01

RAI-02

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	-------------------NOTE------------------- Isolation of SWS flow to individual components does not render the SWS header inoperable. -- Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.9.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.9.3	Verify each SWS pump starts automatically on an actual or simulated actuation signal.	24 months

B 3.7 PLANT SYSTEMS

B 3.7.9 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

The SWS consists of two separate, 100% capacity, safety related, cooling water headers. Each header is supplied by three pumps and includes the piping up to and including the isolation valves on individual components cooled by the SW. Each of the 6 SWS pumps is equipped with rotary strainers and isolation valves.

SWS heat loads are designated as either essential or nonessential. The essential SWS heat loads are those which must be supplied with cooling water immediately in the event of a LOCA and/or loss of offsite power (LOOP). Examples of essential loads are the emergency diesel generators (EDGs), containment fan cooler units (FCUs) and control room air conditioning system (CRACS). The nonessential SWS heat loads are those which are required only following the switch over to the recirculation phase following a postulated LOCA. Examples of nonessential loads are the component cooling water (CCW) heat exchangers.

The FCUs are connected in parallel to the essential SWS header. Normal SWS flow to the FCUs is controlled by TCV-1103. Required ESFAS flow to all five FCUs is initiated when either of the redundant SWS to FCU ESFAS valves (TCV-1104 or TCV-1105) opens automatically in response to an ESFAS actuation signal.

The EDGs are connected in parallel to the essential SWS header. Required ESFAS flow to all three EDGs is initiated when either of the redundant SWS to EDG ESFAS valves (FCV-1176 or FCV-1176A) opens automatically in response to an ESFAS actuation which starts the EDGs.

(continued)

BASES

BACKGROUND (continued)

The CRACS are connected in parallel to the essential SWS header. Required ESFAS flow to both CRACS is provided continuously because the redundant SWS to CRACS valves (TCV-1310/1311 and TCV-1312/1313) have been modified to provide the required flow at all times.

Either of the two SWS headers can be aligned to supply the essential heat loads or the nonessential SWS heat loads. Both the essential and nonessential SWS HEADERS are operated to support normal plant operation and the plant response to accidents and transients. The SWS PUMPS associated with the SWS header designated as the essential header will start automatically. The SWS pumps associated with the SWS header designated as the nonessential header must be manually started when required following a LOCA.

The essential SWS heat loads can be cooled by any two of the three service water pumps on the essential header. The nonessential SWS heat loads can be cooled by any one of the three service water pumps on the nonessential header. To ensure adequate flow to the essential header, the essential and nonessential headers may be cross connected only as necessary while swapping the essential SWS header with the non essential SWS header.

Service water pump suctions are located below the mean sea level in the Hudson River, the ultimate heat sink. This configuration ensures adequate submergence of the SWS pump suctions.

Additional information about the design and operation of the SW, along with a list of the components served, is presented in the FSAR, Section 9.6, (Ref. 1). The principal safety related function of the SWS is the removal of decay heat from the reactor via the CCW System.

APPLICABLE SAFETY ANALYSES

The design basis of the SWS is as follows: post accident essential SWS heat loads can be cooled by any two of the three

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

service water pumps on the designated essential header; and, post accident nonessential SWS heat loads can be cooled by any one of the three service water pumps on the designated nonessential header. With the minimum number of pumps operating, the essential and nonessential headers of the SWS have the required capacity to remove core decay heat following a design basis LOCA as discussed in References 1, 2 and 3. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the ECCS pumps. The Service Water System was designed to fulfill required safety functions while sustaining: (a) the single failure of any active component used during the injection phase of a postulated LOCA with or without a LOOP, or (b) the single failure of any active or passive component used during the long-term recirculation phase with or without a LOOP.

The operating modes of the IP3 SWS are as follows: a) normal mode; b) post-LOCA injection mode; and, c) post-LOCA recirculation mode. The postulated failure conditions of the SWS must include consideration of the limiting case for each operating mode of the system which are as follows:

a. Loss of the 10 inch turbine building service water supply header during normal operation and a seismic event;
b. Loss of instrument air, during the post-LOCA injection phase concurrent with single active component failure.
c. Loss of a SWS pump on both the essential and nonessential headers (resulting from an EDG failure) during the post-LOCA recirculation phase.

The SW, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of CCW and RHR system flow, SWS flow and UHS temperature. The design assumes a maximum SWS temperature of 95°F occurring simultaneously with maximum heat loads on the system (Ref. 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SWS satisfies Criterion 3 of 10 CFR 50.36.

LCO

Three of the three SWS pumps associated with the SWS header designated as the essential header; and, two of the three SWS pumps associated with the SWS header designated as the nonessential header must be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, while sustaining: (a) the single failure of any active component used during the injection phase of a postulated LOCA with or without a LOOP, or (b) the single failure of any active or passive component used during the long-term recirculation phase with or without a LOOP.

An SWS header is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. The required number of pumps, consistent with the header's designation as the essential or nonessential header, are OPERABLE; and

b. The essential and nonessential headers are isolated from each other by at least one closed valve except as specified by the NOTE to the ACTIONS; RAI-01

c. The associated piping, valves, instrumentation and controls required to perform the safety related function are OPERABLE.

The SWS to FCU valves (TCV-1104 or TCV-1105) and SWS to EDG valves (FCV-1176 or FCV-1176A) are OPERABLE when they open automatically in response to ESFAS actuation signal or are blocked open.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS and required to be OPERABLE in these MODES.

(continued)

BASES

APPLICABILITY (continued)

In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports.

ACTIONS

The ACTIONS are modified by a Note that specifies that LCO 3.0.3 is not applicable for 8 hours while swapping the essential SWS header with the nonessential SWS header but only if LCO 3.7.9 will be met after the essential and non-essential header are swapped. This means that the essential and nonessential SWS headers may be cross-connected for up to 8 hours during transfer of the designated essential SWS header to the alternate SWS header. This is acceptable because the transfer is performed infrequently (i.e., approximately every 90 days) and the low probability of an event while the headers are cross connected. RAI-01

A.1 and B.1

If one of the three required SWS pumps on the essential SWS header is inoperable (i.e., Condition A), three Operable pumps must be restored to the essential SWS header within 72 hours. Likewise, if one of the two required SWS pumps on nonessential SWS header is inoperable (i.e., Condition B), the header must be restored so that there are two Operable pumps for the nonessential SWS header within 72 hours. With one required SWS pump inoperable on either or both SWS headers, the remaining OPERABLE SWS pumps are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in an OPERABLE SWS pump could result in loss of SWS function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE pump(s) in the same header, and the low probability of a DBA occurring during this time period. RAI-01

C.1 and D.1

Required ESFAS flow to all three EDGs is initiated when either of the redundant SWS to EDG valves (FCV-1176 or FCV-1176A) opens automatically in response to an ESFAS actuation which starts the RAI-01

(continued)

BASES

ACTIONS

C.1 and D.1 (continued)

EDGs. Similarly, required ESFAS flow to all five FCUs is initiated when either of the redundant SWS to FCU valves (TCV-1104 or TCV-1105) opens automatically in response to an ESFAS actuation signal. The SWS to FCU valves and SWS to EDG valves are OPERABLE when they open automatically in response to an ESFAS actuation signal or are blocked open.

If one of the redundant SWS to EDG valves is inoperable, a single failure of the redundant valve could result in the failure of all three EDGs shortly after the initiation of an event. If one of the redundant SWS to FCU valves is inoperable, a single failure of the redundant valve could result in the failure of all five FCUs. Therefore, a Completion Time of 12 hours is established to restore the required redundancy.

A 12 hour Completion Time is acceptable for the SWS to EDG valves because SWS to the EDGs is still available and the low probability of an event with a loss of offsite power during this period. A 12 hour Completion Time is acceptable for the SWS to FCU valves because SWS to the FCUs is still available, the availability of Containment Spray, and the low probability of an event during this period.

If both SWS to EDG valves or both SWS to FCU valves are inoperable, entry into LCO 3.0.3 is required.

E.1 and E.2

RAI-01

If more than one required SWS pump in either the essential or the nonessential header is inoperable; or, if the flow path associated with either header is not capable of performing its safety function (e.g., both SWS to EDG valves or both SWS to FCU valves are inoperable), then the unit must be placed in a MODE in which the LCO does not apply.

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

Additionally, if an SWS header cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply.

To achieve the required status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the SW.

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 92 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.9.3

This SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.6.
2. FSAR, Section 6.2.
3. WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increase to 95°F at Indian Point 3."

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.9:
"SERVICE WATER SYSTEM"

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
3.3-10	145	145	
3.3-10a	98	98	
3.3-19	145;97-175	145;9-22-98	Change to Bases Page, no impact on ITS
T4.1-3(1)	178;97-156;98-043	200	Amendments 182, 185, and 200 have no impact on ITS 3.7.1

SEE ITS 3.7.8

3. If the Component Cooling System is not restored to meet the requirements of 3.3.E.1 within the time periods specified in 3.3.E.2, then:

a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and in the cold shutdown condition within the following 24 hours.

b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.E.1 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

F. Service Water System ~~Ultimate Heat Sink~~ SEE ITS 3.7.10 A.3

LCO 3.7.9 Applicability 1. ~~The reactor shall not be brought above cold shutdown unless~~ Mode 1, 2, 3 & 4

LCO 3.7.9 a. Three service water pumps on the designated essential header and a minimum of two service water pumps on the designated non-essential header, ~~together with their associated piping and valves~~, are operable. A.1

SEE ITS 3.7.10 b. The service water inlet temperature is less than or equal to 95°F.

Req Act C.1, D.1 / Req Act E.1 / Req Act E.2 2. ~~When the reactor is above cold shutdown and~~ if the requirements of 3.3.F.1.a cannot be met within twelve hours, the reactor shall be brought to the cold shutdown condition, starting no later than the end of the twelve hour period, ~~utilizing normal operating procedures~~ Mode 3 in 6 hrs; Mode 5 in 36 hrs A.3 M.2 A.4 R.1

SEE ITS 3.7.10 3. When the reactor is above cold shutdown and if the requirement of 3.3.F.1.b is exceeded, the reactor shall be placed in at least hot shutdown within seven hours, and in cold shutdown within the following thirty hours unless the service water inlet temperature decreases to within the requirement of 3.3.F.1.b.

~~Add LCO 3.7.9, Actions, Note 1: Separate Cond entry~~ ~~A.5~~ R.1

Add Condition and and associated Req Act. A.1, B.1 M.1 L.1 R.1

Add Condition E and Req Action E.1 M.1 M.2

LA.1

LCO 3.7.9 Note | R.1

4. ~~Isolation shall be maintained between the essential and non-essential headers at all times when above cold shutdown conditions~~ except that for a period of eight hours the headers may be connected while another essential header is being placed in service as described in F.2 above. A.7

SEE ITS 3.7.10

5. At least two service water inlet temperature monitoring instruments (any combination of installed or portable instruments) shall be operable when the reactor is above 350°F and service water inlet temperature exceeds 90°F.

6. If the requirements of 3.3.F.5 cannot be met, the reactor shall be placed in the hot shutdown condition within the next seven hours and subsequently cooled below 350°F using normal operating procedures.

7. Service water inlet temperature shall be the average of two or more service water inlet temperature monitoring instrument readings per 3.3.F.5 taken within a five minute interval (instantaneous).

8. When the reactor is above 350°F and service water inlet temperature per 3.3.F.7 exceeds 90°F, service water inlet temperature monitoring shall commence at a frequency of once per hour.

Add SR 3.7.9.1 M3

Add Note to SR 3.7.9.1 A.6

Add SR 3.7.9.2 M.4

Add SR 3.7.9.3 M.5

ITS/3.7.9 rev 1

~~A total of six service water pumps are installed. Only two of the set of three service water pumps on the header designated the essential header are required immediately following a postulated loss-of-coolant accident.[8] During the recirculation phase of the accident, two service water pumps on the non-essential header will be manually started to supply cooling water for one component cooling system heat exchanger, one control room air conditioner, and one diesel generator; the other component cooling system heat exchanger, the other control room air conditioner, the two other diesel generators and remaining safety related equipment are cooled by the essential service water header.[14] During the recirculation phase of the acccident, both control room air conditioner units may be cooled by the essential service water header.~~

~~The operability requirements on service water temperature monitoring instrumentation and the frequency of service water temperature monitoring insures that appropriate action can be taken to preclude operation beyond established limits. The locations selected for monitoring river water temperature are typically at the circulating or service water inlets, at the circulating water inlet boxes to the condenser hotwells or at the service water supply header to the fan cooler units. Temperature measurements at each of these locations are representative of the river water temperature supplied to cool plant heat loads. Alternate locations may be acceptable on this basis. The limit on the service water maximum inlet temperature insures that the service water and component cooling water systems will be able to dissipate the heat loads generated in the limiting design basis accident[15]. This restriction allows up to seven hours for river water temperature transients which may temporarily increase the service water inlet temperature due to tidal effects to dissipate.~~

~~The operability of the equipment and systems required for the control of hydrogen gas ensures that this equipment is available to maintain the hydrogen concentration within containment below the flammable limit during post-LOCA conditions. Hydrogen concentration exceeding the flammable limit could potentially result in a containment wide hydrogen burn. This could lead to overpressurization of containment, a breach of CONTAINMENT INTEGRITY, containment leakage, unacceptably high offsite doses, and damage to safety-related equipment located in containment. Two full rated recombiner units are provided in order to control the hydrogen evolved in containment following a loss-of-coolant accident. Each unit is capable of preventing the hydrogen concentration from exceeding the flammable limit. Each recombiner is installed such that independence is maintained and redundancy is assured. Each hydrogen recombiner system consists of a recombiner located inside containment, and a separate power supply, and control panel located outside containment such that they are accessible following a design basis accident.~~

A.1

ITS 3.7.9 rev. 1

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS

	Check	Frequency
1. Control Rods	Rod drop times of all control rods	24M
2. Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations
3. Pressurizer Safety Valves	Set Point	24M
4. Main Steam Safety Valves	Set Point	24M
5. Containment Isolation System	Automatic actuation	24M
6. Refueling System Interlocks	Functioning	Each refueling, prior to movement of core components
7. Primary System Leakage	Evaluate	5 days/week
8. Diesel Generators Nos. 31, 32 & 33 Fuel Supply	Fuel Inventory	Weekly
9. Turbine Steam Stop And Control Valves	Closure	Not to exceed 6 months**
10. L.P. Steam Dump System (6 lines)	Closure	Monthly
11. Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Quarterly LA.2
12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	24M

See CTS MASTER MARKUP

LCO 3.7.9

SEE CTS MASTER MARKUP

** The turbine steam stop and control valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probablistic Evaluation of Reduction in Turbine Valve Test Frequency," as updated by Westinghouse Report, WOG-TVTF-93-17, "Update of BB-95/96 Turbine Valve Failure Rates and Effect on Destructive Overspeed Probabilities." The maximum test interval for these valves shall not exceed six months. Surveillance interval extension as per Technical Specification 1.12 is not applicable to the maximum test interval.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.9:
"Service Water System (SWS)"

PART 3:

DISCUSSION OF CHANGES

Differences between CTS and ITS

DISCUSSION OF CHANGES
ITS SECTION 3.7.9 - Service Water System (SWS)

ADMINISTRATIVE

A.1 In the conversion of the Indian Point Unit 3 Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Additionally, editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Rev. 1, i.e., the Improved Standard Technical Specifications.

The CTS Bases are deleted and replaced with comprehensive ITS Bases designed to support interpretation and implementation of the associated Technical Specifications. The Bases explain, clarify, and document the reasons (i.e., bases) for the associated Technical Specifications, and reflect the IP3 plant specific design, analyses, and licensing basis. In accordance with 10 CFR 50.36(a), the ITS Bases are included with the proposed ITS conversion application; however, deletion of the CTS Bases and the adoption of the ITS Bases is an administrative change with no impact on safety.

A.2 CTS Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) include statements of the objective and the applicability. The CTS statements of objective and applicability are deleted because these statements do not establish any requirements and do not provide any guidance for the application of CTS requirements. Therefore, deletion of these statements has no significant adverse impact on safety.

A.3 CTS 3.3.F specifies the Applicability for Service Water System/Ultimate Heat Sink as whenever the reactor is above cold shutdown (i.e., Modes 1, 2, 3 and 4). ITS 3.7.9 and ITS LCO 3.7.10 maintain this Applicability by requiring that the Service Water System and Ultimate Heat Sink be Operable in Modes 1, 2, 3 and 4. This is an administrative change with no impact on safety because there is no change to the CTS Applicability.

A.4 CTS 3.3.F.2 specifies that if requirements for Service Water System Operability are not met and not restored within the specified time, then the reactor must be placed in cold shutdown (Mode 5) utilizing normal operating procedures. Under the same conditions, ITS 3.7.9, Required Actions E.1 and E.2, require that the plant be in Mode 3 in 6 hours and Mode 5 in 36 hours. This change is acceptable because these times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Allowing 36 hours to reach Mode 5 when there is less than the full service water capacity is acceptable because there is a low probability of a DBA occurring during the time allowed to reach Mode 5. This is an administrative change with no impact on safety because it is a reasonable interpretation of existing requirements.

A.5 Not Used.

A.6 CTS 3.3.F does not include any specific requirements or guidance related to the effect on SWS Operability when components or systems supported by SWS are isolated. SR 3.7.9.1 is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable, but does not affect the Operability of the SWS. This is an administrative change with no impact on safety because it is an explicit statement of a reasonable interpretation of the existing requirement.

A.7 CTS 3.3.F.4 specifies that isolation must be maintained between the essential and nonessential headers at all times (See ITS 3.7.9, DOC LA.1) except that for a period of eight hours the headers may be connected while another essential header is being placed in service. ITS LCO 3.7.9, Actions Note, maintains and clarifies this requirement as follows: "If LCO 3.7.9 will be met after the essential and non-essential header are swapped, then LCO 3.0.3 is not applicable for 8 hours while swapping the essential SWS header with the nonessential SWS header." The ITS clarification, if LCO 3.7.9 will be met after the essential and non-essential header are swapped, ensures that the Note is not used to circumvent Service Water system allowable out of service times. This is

an administrative change with no adverse impact on safety because it is a reasonable interpretation of the CTS requirement.

MORE RESTRICTIVE

M.1 CTS 3.3.F.1 requires 3 service water pumps on the essential SWS header because 2 pumps are required in the accident analysis; CTS 3.3.F.1 requires 2 service water pumps on the nonessential SWS header because 1 pump is required in the accident analysis. If one or more pumps on either or both headers are inoperable, CTS 3.3.F.2 allows 12 hours for restoration. Note that CTS 3.3.F.2 permits the same Allowable Out of Service Times (AOTs) for a loss of function or a loss of redundancy on the essential and/or nonessential service water headers.

ITS LCO 3.7.9, Required Actions A.1 (for the essential SW header) and B.1 (for the nonessential header), revise CTS 3.3.F.2 to differentiate between a loss of function and a loss of redundancy on the essential and/or nonessential service water headers when a SW pump is inoperable. ITS LCO 3.7.9, Required Actions A.1 and B.1, establish an AOT of 72 hours (See ITS 3.7.9, DOC L.1) when one inoperable essential and/or nonessential SW pump causes a loss of SW redundancy on the essential and/or nonessential service water headers but both the essential and nonessential SW function are maintained (i.e., at least two essential and one nonessential SW pumps are Operable). If there is a loss of minimum required essential and/or nonessential SW function, ITS LCO 3.7.9, Required Actions E.1 and E.2, requires that the plant is promptly placed in a Mode in which the LCO does not apply. This is a more restrictive change because CTS 3.3.F.2 would allow operation with a loss of minimum required essential and/or nonessential SW function to continue for 12 hours before a plant shutdown is required.

The reasons more restrictive requirements are needed and are acceptable are as follows:

ITS LCO 3.7.9, Condition E and Required Actions E.1 and E.2, are Applicable whenever either SWS header inoperable for reasons other than Conditions A and b (i.e., one essential and/or nonessential SW pump

inoperable), Condition C (i.e., redundant SW to FCU valve inoperable) or Condition D (i.e., redundant SW to EDG valve inoperable). Because all other components associated with the Operability of an SWS header are passive components, in almost all cases entry into Condition D will be the result of multiple inoperable pumps on an SWS header or loss of SW to all FCUs or all EDGs. This places the plant outside the design basis; therefore, prompt plant shutdown is warranted. This more restrictive change does not introduce any operation which is un-analyzed while establishing appropriate requirements when the plant is operating outside of the design basis. Therefore, this change has no adverse impact on safety.

M.2 CTS 3.3.F.1 requires that the required number of essential SW pumps are operable "together with their associated piping and valves." CTS 3.3.F.2 allows 12 hours for restoration of inoperable SW piping and valves. This 12 hour AOT would apply if one or both of the parallel (redundant) automatic valves that supply SW to Fan Cooler Units (FCUs) (TCV-1104 and TCV-1105) and/or supply SW to Emergency Diesel Generators (EDGs) (FCV-1176 and/or FCV-1176A) are inoperable. If both TCV-1104 and TCV-1105 are inoperable, there is a loss of function of all 5 FCUs. If both FCV-1176 and FCV-1176A are inoperable, there is a loss of function of all 3 EDGs. Note that CTS 3.3.F.2 permits the same Allowable Out of Service Time (AOT) for a loss of redundancy on the essential service water header and a loss of SW function to all EDGs and/or FCUs.

ITS LCO 3.7.9, Required Actions C.1 and D.1, revise CTS 3.3.F.2 to differentiate between a loss of function and a loss of redundancy on the essential service water header when the parallel automatic valves that supply SW to FCU and/or SW to EDG are inoperable. ITS LCO 3.7.9, Required Actions C.1 and D.1, maintain the 12 hour AOT when an inoperable valve causes loss of redundancy in the SW flow path to all FCUs and/or all EDGs but an operable flow path remains. If there is a loss of minimum required essential SW function, ITS LCO 3.7.9, Required Actions E.1 and E.2, require that the plant is promptly placed in a Mode in which the LCO does not apply. This is a more restrictive change because CTS 3.3.F.2 would allow operation with a loss of minimum required essential and/or nonessential SW function to continue for 12

hours before a plant shutdown is required.

The reasons more restrictive requirements are needed and are acceptable are as follows:

ITS LCO 3.7.9, Condition E and Required Action E.1 and E.2, are Applicable whenever the essential SW header inoperable for reasons other than Condition C (i.e., redundant SW to FCU valve inoperable) or Condition D (i.e., redundant SW to EDG valve inoperable). Because all other components associated with the Operability of an SWS header are passive components, in almost all cases entry into Condition E will be the result of multiple inoperable pumps on an SWS header or loss of SW to all FCUs or all EDGs. This places the plant outside the design basis; therefore, a prompt plant shutdown is warranted. This more restrictive change does not introduce any operation which is un-analyzed while establishing appropriate requirements when the plant is operating outside of the design basis. Therefore, this change has no adverse impact on safety.

M.3 CTS 3.3 and CTS 4.1 do not require periodic verification that valves capable of being mispositioned are in the correct position. ITS SR 3.7.9.1 establishes a requirement for verification of the correct alignment for manual, power operated, and automatic valves in the SWS System flow paths every 92 days to provide assurance that the proper flow paths will exist for SWS operation. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification of a proper valve lineup at a Frequency consistent with good engineering practice. Therefore, this change has no adverse impact on safety.

M.4 CTS 3.3 and CTS 4.1 do not specifically require periodic verification that SWS valves will actuate to the correct position when required. ITS SR 3.7.9.2 establishes a requirement to verify every 24 months that each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring

periodic verification that automatic SWS valves function as required. Operating experience indicates that the 24 month Frequency is sufficient to provide a high degree of assurance that SWS valves will remain capable of actuating throughout the SR interval. Therefore, this change has no adverse impact on safety.

M.5 CTS 3.3 and CTS 4.1 do not specifically require periodic verification each SWS pump will start automatically when required although CTS Table 4.1-3, Item 11 (as modified by TSCR 98-043), requires a manual pump start at least once per quarter (See ITS 3.7.9, DOC LA.2). ITS SR 3.7.9.3 establishes a requirement to verify every 24 months that each SWS pump starts automatically on an actual or simulated actuation signal. This change is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification automatic SWS valves function as required. Operating experience indicates that the 24 month Frequency is sufficient to provide a high degree of assurance that SWS pumps will remain capable of starting throughout the SR interval. Therefore, this change has no adverse impact on safety.

<u>LESS RESTRICTIVE</u>

L.1 CTS 3.3.F.1 requires 3 service water pumps on the SWS header designated as essential because 2 pumps are required in the accident analysis; CTS 3.3.F.1 requires 2 service water pumps on the SWS header designated as nonessential because 1 pump is required in the accident analysis. If one or more required pumps on either or both headers are inoperable, CTS 3.3.F.2 allows 12 hours for restoration. Note that CTS 3.3.F.2 permits the same Allowable Out of Service Times (AOTs) for a loss of function or a loss of redundancy on one or both headers.

Under the same conditions, ITS LCO 3.7.9, Required Actions A.1 (for the essential SW header) and B.1 (for the nonessential SW header), revise CTS 3.3.F.2 to provide a less restrictive AOT (72 hours versus 12 hours) when an inoperable essential and/or nonessential SW pump results in a loss of redundancy but functional capability is maintained.

This change is needed to provide an allowable out of service time (AOT) commensurate with the level of degradation resulting from the inoperability of one of the three 50% capacity pumps on the essential SWS header and one of the two 100% capacity pumps on the nonessential SWS header. This change is acceptable because of the following: the remaining Operable pumps are capable of removing the post accident heat load; and, the low probability of an event during the AOT for a pump on the essential and/or nonessential header. This change is supported by ITS LCO 3.8.1, Required Actions, which limits the time that a required component may be inoperable if the normal or emergency power supply to the redundant component is inoperable. Therefore, this change has no significant impact on safety.

REMOVED DETAIL

LA.1 CTS 3.3.F.4 specifies that isolation must be maintained between the essential and nonessential headers at all times when above cold shutdown except for a period of eight hours when the headers may be cross connected while another essential header is being placed in service. The allowance permitting the essential and nonessential headers to be cross connected for 8 hours while changing the SWS lineup is maintained in ITS LCO 3.7.9.1, Actions Note; however, the statement that the SWS essential and nonessential headers must otherwise be isolated is moved to the Bases as a requirement of SWS Operability. The requirement to separate SWS headers is not changed and is still enforced indirectly by SR 3.7.9.1 and the description of requirements for Operability in the ITS Bases. This is acceptable because this valve lineup information is incorporated into the minimum requirements and ITS specifies the minimum requirements for Operability. Therefore, this design information can be adequately defined and controlled in the ITS 3.7.9 Bases which require change control in accordance with ITS 5.5.12, Bases Control Program.

This change, which allows the detailed description of the requirements for Operability of these systems to be maintained in the ITS Bases, is consistent with the approach used in NUREG-1431 for all Limiting Conditions for Operation (LCOs). This approach is acceptable because the requirements of ITS 5.5.13, Technical Specifications (TS) Bases

Control Program, is designed to assure that changes to the ITS Bases do not result in changes to the Technical Specification requirements and do not result in significant increases in the probability or consequences of accidents previously evaluated, do not create the possibility of a new or different kind of accident, and do not result in a significant reduction in a margin of safety. Additionally, IP3 programs that implement ITS Bases changes in accordance with ITS 5.5.13 require periodic submittal of Bases changes to the NRC for review.

This change is a less restrictive administrative change with no impact on safety because no requirements are being deleted from Technical Specifications and an appropriate change control process and an appropriate level of regulatory oversight is maintained for the information being relocated out of the Technical Specifications.

LA.2 CTS Table 4.1-3, Item 11, requires a manual pump start and 15 minutes of pump operation at least quarterly (Amendment 185). ITS SR 3.7.9.3 establishes a new requirement to verify every 24 months that each SWS pump starts automatically on an actual or simulated actuation signal (See ITS 3.7.9, DOC M.4); however, the requirement for a manual pump start and 15 minutes of pump operation at least once per quarter is relocated to the Inservice Testing (IST) Program.

This change is acceptable for the following reasons: the requirement to operate each pump for 15 minutes will be maintained in the IST Program; ITS LCO 3.7.9 maintains the requirement that SWS is Operable; at least 2 of the 6 SWS pumps are running during normal plant operation and the running pumps are rotated; and, operating experience indicates that SWS pumps will remain capable of starting throughout the 24 month SR interval in SR 3.7.9.3.

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. Therefore, maintaining the requirement that SWS pumps must be Operable in ITS 3. 7.9 and maintaining the requirement for periodic testing of pumps in the IST Program required by ITS 5.5.7 provides a high degree of assurance that the SWS will be tested and maintained to ensure SWS 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 ECCS pumps can be maintained in the ITS with the Frequency in the IST program with no significant adverse impact on safety.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.9:
"Service Water System (SWS)"

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 3.7.9 - Service Water System (SWS)

LESS RESTRICTIVE
("L.1" Labeled Comments/Discussions)

New York Power Authority has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria set forth in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change modifies the allowable out of service (AOT) for inoperable service water system (SWS) pumps to differentiate between loss of function and loss of redundancy. This change extends the allowable out of service time for an inoperable SWS pump that does not result in a loss of function from 12 hours to 72 hours. This change will not result in a significant increase in the probability of an accident previously evaluated because the status of SWS pumps has no affect on the initiators of any accident previously evaluated. This change will not result in a significant increase in the consequences of an accident previously evaluated because the remaining Operable pumps are capable of removing the post accident heat load, the amount of time in this condition is limited, and the low probability of an event. This change is supported by ITS LCO 3.8.1, Required Actions, which limits the time that a required component may be inoperable if the normal or emergency power supply to the redundant component is inoperable.

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

   The proposed change will not involve any physical changes to systems, structures, or components, or involve a change in normal plant operation. Therefore, it will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety because the remaining Operable pumps are capable of removing the

post accident heat load, the amount of time in this condition is limited, and the low probability of an event. This change is supported by ITS LCO 3.8.1, Required Actions, which limits the time that a required component may be inoperable if the normal or emergency power supply to the redundant component is inoperable.

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.9:
"Service Water System (SWS)"

PART 5:

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

Status of NUREG 1431 Generic Changes for ITS 3.7.9

This ITS Specification is based on NUREG-1431 Specification No. **3.7.8**
as modified by the following Generic Changes:

OG No.	TSTF No.	Generic Change Description	NRC STATUS	IP3 STATUS	JD No.
N/A	N/A	NO GENERIC CHANGES ARE POSTED AGAINST THIS SPECIFICATION.	Not Applicable	Not Applicable	N/A

<CTS>

3.7 PLANT SYSTEMS

3.7.8 9 Service Water System (SWS)

<3.3.F.1.a> LCO 3.7.8 ~~Two SWS trains shall be OPERABLE.~~ Insert: 3.7-19-01

APPLICABILITY: MODES 1, 2, 3, and 4. Insert: 3.7-19-02

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One SWS train inoperable.~~	~~A.1 --------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for residual heat removal loops made inoperable by SWS. ------------------------ Restore SWS train to OPERABLE status.~~	~~72 hours~~

<3.3.F.2>
<DOC L.1>
<DOC M.1>

Insert: 3.7-19-03

Insert: 3.7-19-04

(continued)

NUREG-1431 Markup Inserts
ITS SECTION 3.7.9 - Service Water System (SWS)

INSERT: 3.7-19-01

<3.3.F.1.a> Three pumps and required flow path for the essential SWS header shall be Operable;

AND,

<3.3.F.1.a> Two pumps and required flow path for the nonessential SWS header shall be Operable.

INSERT: 3.7-19-02

<3.3.F.4>
<DOC A.7>

-----NOTE-----
If LCO 3.7.9 will be met after the essential and non-essential header are swapped, then LCO 3.0.3 is not applicable for 8 hours while swapping the essential SWS header with the nonessential SWS header. R.1

INSERT: 3.7-19-03

<3.3.F.2> <DOC M.1> <DOC L.1>	A.	One required SW pump on essential header inoperable.	A.1	Establish 3 OPERABLE SW pumps on the essential SW header.	72 hours	R.1
<3.3.F.2> <DOC M.1> <DOC L.1>	B.	One required SW pump on nonessential header inoperable.	B.1	Establish 2 OPERABLE SW pumps on the nonessential SW header.	72 hours	R.1

NUREG-1431 Markup Inserts
ITS SECTION 3.7.9 - Service Water System (SWS)

INSERT: 3.7-19-04

<3.3.F2> <DOC M.2> <DOC L.1>	C.	One EDG ESFAS Service Water valve inoperable.	C.1	Restore both EDG ESFAS Service Water valves to OPERABLE status.	12 hours	R.1
<3.3.F2> <DOC L.1> <DOC M.2>	D.	One FCU ESFAS Service Water valve inoperable.	D.1	Restore both FCU ESFAS Service Water valves to OPERABLE status.	12 hours	R.1

ACTIONS (continued)

<3.3.F.2>
<DOC A.4>

CONDITION	REQUIRED ACTION		COMPLETION TIME
E ~~B~~. Required Action and associated Completion Time of Condition A B, C or D not met.	E ~~B~~.1	Be in MODE 3.	6 hours
	AND		
	E ~~B~~.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<DOC A.6> <DOC M.3>	SR 3.7.~~8~~.1 9	-------------------NOTE------------------- Isolation of SWS flow to individual components does not render the SWS inoperable. -- Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	header 31 days 92 X.1 | R.1
<DOC M.4>	SR 3.7.~~8~~.2 9	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	[18] months 24
<DOC M.5> <T 4.1-3, #11> <DOC LA.2>	SR 3.7.~~8~~.3 9	Verify each SWS pump starts automatically on an actual or simulated actuation signal.	[18] months 24

B 3.7 PLANT SYSTEMS

B 3.7.~~8~~ 9 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

Insert: B3.7-41-01

~~The SWS consists of two separate, 100% capacity, safety related, cooling water trains. Each train consists of two 100% capacity pumps, one component cooling water (CCW) heat exchanger, piping, valving, instrumentation, and two cyclone separators. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions. The SWS also provides emergency makeup to the spent fuel pool and CCW System [and is the backup water supply to the Auxiliary Feedwater System].~~

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the FSAR, Section ~~[9.2.1]~~ 9.6 (Ref. 1). The principal safety related function of the SWS is the removal of decay heat from the reactor via the CCW System.

APPLICABLE SAFETY ANALYSES

Insert: B3.7-43-02

~~The design basis of the SWS is for one SWS train, in conjunction with the CCW System and a 100% capacity containment cooling system, to remove core decay heat following a design basis LOCA as discussed in the FSAR, Section [6.2] (Ref. 2).~~ This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the ECCS pumps.

Insert: B3.7-43-03

~~The SWS is designed to perform its function with a single failure of any active component, assuming the loss of offsite power.~~

~~(continued)~~

NUREG-1431 Markup Inserts
ITS SECTION 3.7.9 - Service Water System (SWS)

Insert: B 3.7-41-01 (page 1 of 2)

The SWS consists of two separate, 100% capacity, safety related, cooling water headers. Each header is supplied by three pumps and includes the piping up to and including the isolation valves on individual components cooled by the SWS. Each of the 6 SWS pumps is equipped with rotary strainers and isolation valves.

SWS heat loads are designated as either essential or nonessential. The essential SWS heat loads are those which must be supplied with cooling water immediately in the event of a LOCA and/or loss of offsite power (LOOP). Examples of essential loads are the emergency diesel generators (EDGs), containment fan cooler units (FCUs) and control room air conditioning system (CRACS). The nonessential SWS heat loads are those which are required only following the switch over to the recirculation phase following a postulated LOCA. Examples of nonessential loads are the component cooling water (CCW) heat exchangers.

The FCUs are connected in parallel to the essential SWS header. Normal SW flow to the FCUs is controlled by TCV-1103. Required ESFAS flow to all five FCUs is initiated when either of the redundant SW to FCU ESFAS valves (TCV-1104 or TCV-1105) opens automatically in response to an ESFAS actuation signal.

The EDGs are connected in parallel to the essential SWS header. Normal Required ESFAS flow to all three EDGs is initiated when either of the redundant SW to EDG ESFAS valves (FCV-1176 or FCV-1176A) opens automatically in response to an ESFAS actuation which starts the EDGs.

The CRACs are connected in parallel to the essential SWS header. Required ESFAS flow to both CRACS is provided continuously because the redundant SW to CRACS valves (TCV-1310/1311 and TCV-1312/1313) have been modified to provide the required flow at all times.

Insert: B 3.7-41-01 (page 2 of 2)

Either of the two SWS headers can be aligned to supply the essential heat loads or the nonessential SWS heat loads. Both the essential and nonessential SWS headers are operated to support normal plant operation and the plant response to accidents and transients. The SWS pumps associated with the SWS header designated as the essential header will start automatically. The SWS pumps associated with the SWS header designated as the nonessential header must be manually started when required following a LOCA.

The essential SWS heat loads can be cooled by any two of the three service water pumps on the essential header. The nonessential SWS heat loads can be cooled by any one of the three service water pumps on the nonessential header. To ensure adequate flow to the essential header, the essential and nonessential headers may be cross connected only as necessary while swapping the essential SWS header with the nonessential SWS header.

Service water pump suctions are located below the mean sea level in the Hudson River, the ultimate heat sink. This configuration ensures adequate submergence of the SWS pump suctions.

NUREG-1431 Markup Inserts
ITS SECTION 3.7.9 - Service Water System (SWS)

Insert: B 3.7-41-02

The design basis of the SWS is as follows: post accident essential SWS heat loads can be cooled by any two of the three service water pumps on the designated essential header; and, post accident nonessential SWS heat loads can be cooled by any one of the three service water pumps on the designated nonessential header. With the minimum number of pumps operating, the essential and nonessential headers of the SWS have the required capacity to remove core decay heat following a design basis LOCA as discussed in References 1, 2 and 3.

Insert: B 3.7-41-03

The Service Water System was designed to fulfill required safety functions while sustaining: (a) the single failure of any active component used during the injection phase of a postulated LOCA with or without a LOOP, or (b) the single failure of any active or passive component used during the long-term recirculation phase with or without a LOOP.

The operating modes of the IP3 SWS are as follows: a) normal mode; b) post-LOCA injection mode; and, c) post-LOCA recirculation mode. The postulated failure conditions of the SWS must include consideration of the limiting case for each operating mode of the system which are as follows:

a. Loss of the 10 inch turbine building service water supply header during normal operation and a seismic event;
b. Loss of instrument air, during the post-LOCA injection phase concurrent with single active component failure.
c. Loss of a SWS pump on both the essential and nonessential headers (resulting from an EDG failure) during the post-LOCA recirculation phase.

BASES

APPLICABLE SAFETY ANALYSES (continued)

flow, SWS flow and UHS temperature

The SWS, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR)~~, as discussed in the FSAR, Section [5.4.7], (Ref. 3)~~ entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of ~~the number of~~ CCW and RHR System ~~trains that are operating~~. ~~One SWS train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6.~~ The design ~~This~~ assumes a maximum SWS temperature of ~~[~~95~~]~~°F occurring simultaneously with maximum heat loads on the system. (Ref. 3)

10 CFR 50.36

The SWS satisfies Criterion 3 of ~~the NRC Policy Statement~~.

LCO

Insert: B3.7-42-01

~~Two SWS trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.~~

header

An SWS ~~train~~ is considered OPERABLE during MODES 1, 2, 3, and 4 when:

Insert: B3.7-42-02

a. ~~The pump is OPERABLE;~~ ~~and~~

Insert: B3.7-42-03

c ~~b~~. The associated piping, valves, ~~heat exchanger, and~~ instrumentation and controls required to perform the safety related function are OPERABLE.

Insert: B3.7-42-04

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports.

ACTIONS

Insert: B3.7-42-05

A.1 and B.1 | R.1

Insert: B3.7-42-06

~~If one SWS train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition~~

~~(continued)~~

NUREG-1431 Markup Inserts
ITS SECTION 3.7.9 - Service Water System (SWS)

Insert: B 3.7-42-01

Three of the three SWS pumps associated with the SWS header designated as the essential header; and, two of the three SWS pumps associated with the SWS header designated as the nonessential header must be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, while sustaining: (a) the single failure of any active component used during the injection phase of a postulated LOCA with or without a LOOP, or (b) the single failure of any active or passive component used during the long-term recirculation phase with or without a LOOP.

Insert: B 3.7-42-02

The required number of pumps, consistent with the header's designation as the essential or nonessential header, are OPERABLE;

Insert: B 3.7-42-03

b. The essential and nonessential headers are isolated from each other by at least one closed valve except as specified by the NOTE to the ACTIONS;

Insert: B 3.7-42-04

The SW to FCU valves (TCV-1104 or TCV-1105) and SW to EDG valves (FCV-1176 or FCV-1176A) are OPERABLE when they open automatically in response to an ESFAS actuation signal or are blocked open.

NUREG-1431 Markup Inserts
ITS SECTION 3.7.9 - Service Water System (SWS)

Insert: B 3.7-42-05

The ACTIONS are modified by a Note that specifies that LCO 3.0.3 is not applicable for 8 hours while swapping the essential SWS header with the nonessential SWS header but only if LCO 3.7.9 will be met after the essential and non-essential header are swapped. This means that the essential and nonessential SWS headers may be cross-connected for up to 8 hours during transfer of the designated essential SWS header to the alternate SWS header. This is acceptable because the transfer is performed infrequently (i.e., approximately every 90 days) and the low probability of an event while the headers are cross connected. | R.1

Insert: B 3.7-42-06

If one of the three required SWS pumps on the essential SWS header is inoperable (i.e., Condition A), three Operable pumps must be restored to the essential SWS header within 72 hours. Likewise, if one of the two required SWS pumps on nonessential SWS header is inoperable (i.e., Condition B), the header must be restored so that there are two Operable pumps for the nonessential SWS header within 72 hours. With one required SWS pump inoperable on either or both SWS headers, | R.1 | R.1

BASES

ACTIONS

A.1 and B.1 (continued)

the remaining OPERABLE SWS ~~train is~~ pumps are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in ~~the~~ an OPERABLE SWS ~~train~~ pump could result in loss of SWS function. ~~Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," should be entered if an inoperable SWS train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," should be entered if an inoperable SWS train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.~~ The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE ~~train~~ pump(s) in the same header, and the low probability of a DBA occurring during this time period.

Insert: B3.7-43-01

~~B~~E.1 and ~~B~~E.2 R.1

~~If~~ H the SWS ~~train~~ header cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve ~~this~~ required status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

Insert: B3.7-43-02

SURVEILLANCE REQUIREMENTS

SR 3.7.~~8~~9.1

This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the SWS.

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or

~~(continued)~~

NUREG-1431 Markup Inserts
ITS SECTION 3.7.9 - Service Water System (SWS)

Insert: B 3.7-43-01

~~B~~C.1 and ~~C~~D.1 R1

Required ESFAS flow to all three EDGs is initiated when either of the redundant SW to EDG ESFAS valves (FCV-1176 or FCV-1176A) opens automatically in response to an ESFAS actuation which starts the EDGs. Similarly, required ESFAS flow to all five FCUs is initiated when either of the redundant SW to FCU ESFAS valves (TCV-1104 or TCV-1105) opens automatically in response to an ESFAS actuation signal. The SW to FCU ESFAS valves and SW to EDG ESFAS valves are OPERABLE when they open automatically in response to an ESFAS actuation signal or are blocked open.

If one of the redundant SW to EDG ESFAS valves is inoperable, a single failure of the redundant valve could result in the failure of all three EDGs shortly after the initiation of an event. If one of the redundant SW to FCU ESFAS valves is inoperable, a single failure of the redundant valve could result in the failure of all five FCUs. Therefore, a Completion Time of 12 hours is established to restore the required redundancy.

A 12 hour Completion Time is acceptable for the SW to EDG valves because SW to the EDGs is still available and the low probability of an event with a loss of offsite power during this period. A 12 hour Completion Time is acceptable for the SW to FCU valves because SW to the FCUs is still available, the availability of Containment Spray, and the low probability of an event during this period.

If both SW to EDG valves or both SW to FCU valves are inoperable, entry into LCO 3.0.3 is required.

Insert: B 3.7-43-02

If more than one required SWS pump in either the essential or the nonessential header is inoperable; or, the flow path associated with either header is not capable of performing its safety function (e.g., both SW to EDG valves or both SW to FCU valves are inoperable), then the plant in a MODE in which the LCO does not apply. Additionally, if

BASES

SURVEILLANCE REQUIREMENTS

9

SR 3.7.~~8~~.1 (continued)

otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

92

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

9

SR 3.7.~~8~~.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

24

9

SR 3.7.~~8~~.3

This SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month

24

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~8~~.3 (continued) 9

Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section ~~[9.2.1]~~. 9.6
2. FSAR, Section ~~[6.2]~~. 6.2
3. ~~FSAR, Section [5.4.7]~~.

Insert: B 3.7-45-01

Insert: B 3.7-45-01

3. WCAP-12313, "Safety Evaluation for an Ultimate Heat Sink Temperature Increase to 95°F at Indian Point 3."

Indian Point 3
Improved Technical Specifications (ITS)
Conversion Package

Technical Specification 3.7.9:
"Service Water System (SWS)"

PART 6:

Justification of Differences between

NUREG-1431 and IP3 ITS

JUSTIFICATION OF DIFFERENCES FROM NUREG-1431
ITS SECTION 3.7.9 - Service Water System (SWS)

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. There are no technical changes to requirements as specified in NUREG 1431, Rev 1; therefore, this change is not a significant or generic deviation from NUREG 1431, Rev 1.

DB.2 NUREG-1431, WOG STS Section 3.7.7, Required Action A.1, is modified by two Notes indicating that the applicable Conditions and Required Actions of LCO 3.8.1, AC Sources - Operating, and/or LCO 3.4.6, RCS Loops – Mode 4, must be entered if an inoperable SWS header results in an inoperable EDG and/or RHR loop. ITS 3.7.9 does not have these notes nor does it allow a Condition for an inoperable service water header. IP3 SWS design has three required SWS pumps on the essential header and two required SWS pumps on the non-essential header. The essential header supports EDGs and the non-essential header supports RHR. This design is such that when one required SWS pump is inoperable in one or both of the

two headers, then the SWS support function for the EDG and RHR systems are met by the remaining required pump(s) on the headers. ITS 3.7.9 allows only one required SWS pump for each header to be inoperable. Hence, ITS 3.7.9 Condition statements only allow a reduction in redundancy while the SWS support function is maintained for the EDG and RHR trains without the need for the Notes that modified the Actions in the STS. Since, the above described difference as presented by ITS 3.7.9 does not mitigate the supported EDG and RHR systems functions for the applicable modes of operation, there is no adverse impact on safety.

DIFFERENCE BASED ON A GENERIC CHANGE TRAVELER FOR NUREG-1431

None

DIFFERENCES FOR ANY REASON OTHER THAN ABOVE

X.1 IP3 CTS has no requirement to periodically verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or other wise secured in position, is in the correct position. IP3 is voluntarily adopting this STS SR 3.7.9.1, at a frequency of 92 days versus the 31 day frequency in NUREG-1431. The 31 day frequency would be a significant burden without commensurate contribution to plant safety. The 92 day frequency is consistent with the current frequency for pump tests that includes verifying proper valve position during the header swap and operating experience has shown the current frequency to be adequate. This frequency is acceptable because it does not introduce any operation which is un-analyzed while requiring periodic verification of proper valve lineup at a frequency consistent with good practice.