

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

**MARKUP OF CURRENT TECHNICAL SPECIFICATIONS
(CTS)**

DISCUSSION OF CHANGES (DOCs) TO THE CTS

**NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
FOR LESS RESTRICTIVE CHANGES**

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

**JUSTIFICATION FOR DIFFERENCES (JFDs) FROM
NUREG-1433, REVISION 1**

MARKUP OF NUREG-1433, REVISION 1, BASES

**JUSTIFICATION FOR DIFFERENCES (JFDs) FROM
NUREG-1433, REVISION 1, BASES**

**RETYPE PROPOSED IMPROVED TECHNICAL
SPECIFICATIONS (ITS) AND BASES**

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

**MARKUP OF CURRENT TECHNICAL
SPECIFICATIONS (CTS)**

AI

3.3.B.3 (cont'd)

4.3.B (cont'd)

c. When required by Specifications 3.3.B.3.a or b, the second licensed reactor operator, licensed senior operator, or the reactor engineer must be present at the reactor console during rod movements to verify compliance with the prescribed rod pattern. This individual shall have no other concurrent duties during the rod withdrawal or insertion.

See ITS: 3.3.2.1

d. Plant startup under Specification 3.3.B.3.b is only permitted once per calendar year. Any startup conducted without the RWM as described in Specification 3.3.B.3.b shall be reported to the NRC within 30 days of the startup. This special report shall state the reason for the RWM inoperability, the action taken to restore it, and the schedule for returning the RWM to an operable status.

See ITS: 3.3.2.1

e. Control rod patterns shall be equivalent to those prescribed by the Banked Position Withdrawal Sequence (BPWS) such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/cm.

LAI

LI

See ITS: 3.3.2.1

add ACTION A

If Specifications 3.3.B.3.a through e cannot be met, the reactor shall not be restarted, or if the reactor is in the run or startup modes at less than 10% rated thermal power, no rod movement is permitted, except by scram.

add Required Action B, 2 and associated completion time

LI

add Required Action B.1 Note

LI

Add SR 3.1.6.1

m2

ECO 3.1.6

A2

ACTION B

Applicability

Required Action B.1

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

**DISCUSSION OF CHANGES (DOCs) TO THE
CTS**

DISCUSSION OF CHANGES
ITS: 3.1.6 - ROD PATTERN CONTROL

ADMINISTRATIVE CHANGES

- A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).
- A2 CTS 3.3.B.3.f in part requires that if the requirements of BPWS can not be met, "the reactor shall not be restarted." In the proposed ITS, the ability to change MODES is generically controlled by the provisions of LCO 3.0.4 which states in part that "when an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time." ITS 3.1.6 Required Action B.2 requires that the plant be shutdown by placing the reactor mode switch in shutdown, and therefore LCO 3.0.4 would prevent plant startup with BPWS requirements not met. Therefore, this proposed change causes no technical or actual change from present specifications. Therefore, the change is considered administrative, and is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 3.3.B.3.f requires the control rod patterns to be equivalent to those prescribed by the BPWS "if the reactor is in the run or startup mode at less than 10% rated thermal power". The Applicability for ITS 3.1.6 requires that Operable control rods comply with the requirements of BPWS in "MODES 1 and 2 with THERMAL POWER \leq 10% RTP". Thus the new Applicability requires the control rod pattern to comply with BPWS at 10% RTP, whereas the CTS does not. Although technically a more restrictive change, this proposed change does not alter plant operations or actions that would be taken if control rods were found outside the control rod pattern governed by BPWS. This change is consistent with NUREG-1433, Revision 1.
- M2 CTS 3.3.B.3.e and f contain the present requirements for control rod patterns to be equivalent to the Banked Position Withdrawal Sequence (BPWS). The ITS proposes to add an SR to verify all Operable control rods comply with BPWS every 24 hours. This new SR is consistent with

DISCUSSION OF CHANGES
ITS: 3.1.6 - ROD PATTERN CONTROL

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 (continued)

NUREG-1433, Revision 1, and represents a new and therefore, more restrictive requirement necessary to ensure the control rod pattern is in accordance with BPWS.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 3.3.B.3.e provides the reason for control rod patterns being equivalent to those prescribed by the BPWS (i.e., such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm). This detail is proposed to be relocated to the Bases for the proposed ITS Specification (3.1.6). These details are not necessary to ensure that the control rods comply with the requirements of the BPWS. The requirement in ITS 3.1.6 ACTION B to limit the number of OPERABLE control rods not in compliance with BPWS to 8 is sufficient to ensure the peak fuel enthalpy limit is not exceeded. Therefore, these relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.3.B.3.f requires all rod movement to be stopped except by scram if control rod patterns and sequence of withdrawal or insertion are not established such that the control rod drop accident limit of 280 cal/g is not exceeded. ITS 3.1.6 ACTION A requires associated control rod(s) to be moved to the correct position or declared inoperable within 8 hours if one or more control rods is not in compliance with BPWS. Non-compliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to $\leq 10\%$ RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to 8, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. Any of the 8 control rods that cannot be restored to its correct position within 8 hours must then be declared inoperable and fully inserted within 3 hours as required by ITS 3.1.3, ACTION C. The time allowed by ITS 3.1.6 to restore out-of-sequence OPERABLE control rods is acceptable because: it is expected that the control rod pattern could be restored to compliance with BPWS in a brief period of time; each control rod

DISCUSSION OF CHANGES
ITS: 3.1.6 - ROD PATTERN CONTROL

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 (continued)

moved during the correction process would require the RWM to be bypassed (which in turn would require a second verification of the proper selection and movement of the control rod); the time allowed for correction is brief with each step bringing the pattern closer to compliance with BPWS; and, the probability of a CRDA during this brief period is remote.

ITS ACTION B requires that withdrawal of control rods be immediately suspended and the reactor mode switch be placed in the shutdown position within 1 hour if nine or more OPERABLE control rods are not in compliance with BPWS. If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal must be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals. ITS 3.1.6 Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. ITS 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff with the RWM bypassed. When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, this action places the plant outside the Applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence. The requirement to place the mode switch in shutdown may seem to be more restrictive, but is necessary since with nine or more control rods not in compliance with BPWS there is a significant departure from the control rod sequence. This change is considered less restrictive since allowances are provided to restore the control rod pattern which is not permitted by the CTS.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

**NO SIGNIFICANT HAZARDS CONSIDERATION
(NSHC) FOR LESS RESTRICTIVE CHANGES**

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.6 - ROD PATTERN CONTROL

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. 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?

The proposed change revises the Required Actions for control rods not in compliance with banked position withdrawal sequence (BPWS) below 10% of rated thermal power (RTP). Control rods not in compliance with BPWS are not in themselves considered as initiators for any accidents previously evaluated and therefore cannot increase the probability of such accidents. The current BPWS generic analysis evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number and distribution of fully inserted, inoperable control rods. Therefore, this change will not contribute to an increase in the consequences of previously evaluated accidents. Additionally, the extended time for ACTION does not affect the ability of the systems to respond to such accidents and also do not contribute to a significant increase in the consequences of an accident previously evaluated. Therefore, no significant increase in the probability of an accident previously evaluated is involved with this change.

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

This change will not physically alter the plant (no new or different types of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change allows additional time to correct control rod patterns which may not be as analyzed. However, these conditions occur infrequently and any minor decrease in the margin during this additional time is offset by not inducing core transients while in this condition.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.6 - ROD PATTERN CONTROL

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

3. (continued)

Therefore, the change does not involve a significant reduction in a margin of safety.

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

**MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION**

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the ~~banked~~ position withdrawal sequence (BPWS) ~~DBI~~

[3.3.B.3e]

DBI

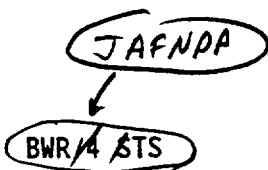
APPLICABILITY: MODES 1 and 2 with THERMAL POWER \leq 10% RTP.

[3.3.B.3f]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[L1] A. One or more OPERABLE control rods not in compliance with BPWS	A.1 -----NOTE----- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." ----- Move associated control rod(s) to correct position.	8 hours
	OR A.2 Declare associated control rod(s) inoperable.	8 hours

(continued)



Amendment
Rev 1, 04/07/95

Typ
All
Pages

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Nine or more OPERABLE control rods not in compliance with (BPWS) DBI</p> <p>[3.3.B3.F] [L1]</p>	<p>B.1 -----NOTE----- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1. -----</p> <p>Suspend withdrawal of control rods.</p> <p>AND</p> <p>B.2 Place the reactor mode switch in the shutdown position.</p>	<p>Immediately</p> <p>1 hour</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>[M2] SR 3.1.6.1 Verify all OPERABLE control rods comply with (BPWS) DBI</p>	<p>24 hours</p>

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

**JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1**

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.6 - ROD PATTERN CONTROL

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the plant specific values or sequence included. As indicated in the Bases, JAFNPP uses the Reduced Notch Worth Procedure (RNWP) which was developed to reduce notch worth even further than the BPWS. This change is considered to comply with the requirement of BPWS.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

DB3

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO₂ have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1 and 2) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 6) and the calculated offsite doses will be well within the required limits (Ref. 5).

PAB
energy deposition

6, 7, 8

DB1

9

10

Insert ASAI

DB1

(continued)

BWR/A STS
JAFNAP

Rev. 04/07/95

Revision 0

Typ All Pages

DBI

INSERT ASA1

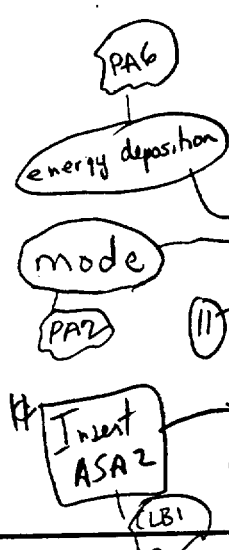
The calculated offsite doses remain within the limits since only a small number of fuel rods would reach a fuel enthalpy of 170 cal/gm, which is the enthalpy limit for eventual cladding perforation.

limit

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 110% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS ~~MODE~~ of operation. The generic BPWS analysis (Ref. 1) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.



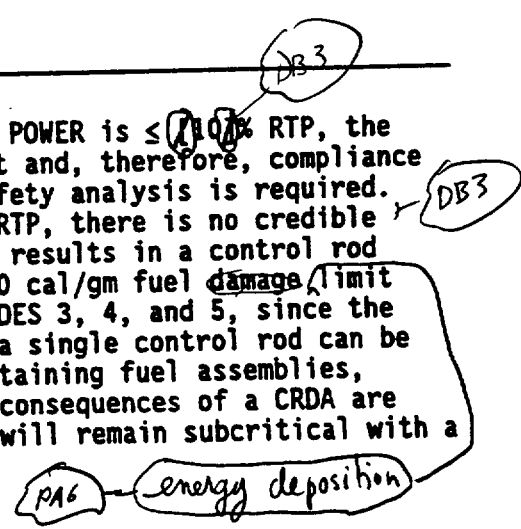
Rod pattern control satisfies Criterion 3 of the NRC Policy Statement. 10 CFR 50.36(c)(2)(ii) (Ref. 13) XI

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is $\leq 110\%$ RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is $> 110\%$ RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.



(continued)

Insert ASA2

CLB1

The current control rod withdrawal sequence utilized at JAFNPP is known as the Reduced Notch Worth Procedure (RNWP) which was developed to reduce notch worth even further than the BPWS (Ref. 12). The CRDA analyses of References 1, 6, 7, 8, 9 and 11 bound the consequences of a CRDA for these plants following RNWP (Ref. 2).

Insert Page B 3.1-35

REVISION D

BASES (continued)

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to $\leq 70\%$ RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the rod pattern, or scram if warranted.

DB3

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

second licensed operator (Reactor Operator or Senior Reactor Operator) or reactor engineer

PA1

RA1 3.1-06

RA1 3.1-07

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff.

Second licensed operator (Reactor Operator or Senior Reactor Operator) or
PAI

Reactor engineer PAI

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at $\leq 10\%$ RTP.

DB3

REFERENCES

1. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, Supplement for United States, Section 2.2.3.1, September 1988. August 1996
 2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.
 3. NUREG-0979, Section 4.2.1.3.2, April 1983.
 4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.
- Safety Evaluation Report Related to the Final Design Approval of the GESSAR II, BWR16 Nuclear Island Design (and Supplements 1 through 5)

DB2
Letter from T.A. Pickens (BWR06) to G.C. Laines (NRC), Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, BWR06-8644, August 15, 1986.

Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants
Spectrum of Rod Prop Accidents (RWR)
PA7
PA7
(continued)

BASES

REFERENCES
(continued)

Insert
Ref 1.

DBI

- 5. 10 CFR 100.11.
- 9. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.
- 10. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda Winter of 1966.
- 11. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

DBI

PAS

NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

DBI

ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda Winter of 1966.

PAS

NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

Insert Ref 2

CCBI

Insert Ref 1

DBI

6. NEDO-10527, Rod Drop Accident Analysis For Large BWRs, March 1972.
7. NEDO-10527, Supplement 1, Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 1, Multiple Enrichment Cores With Axial Gadolinium, July 1972.
8. NEDO-10527, Supplement 2, Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 2, Exposed Cores, January 1973.

Insert Ref 2

CLBI

12. SIL-316, Reduced Notch Worth Procedure, November 1979.
13. 10 CFR 50.36(c)(2)(ii).

XI

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

**JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1, BASES**

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.6 - ROD PATTERN CONTROL

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 Changes have been made to identify the current control rod sequence used at JAFNPP to satisfy the requirements of the generic CRDA safety analysis. The associated References have been added as well.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Changes have been made consistent with the Specification.

PA2 Changes have been made to correct a typographical error.

PA3 Not used

PA4 Not used

PA5 The quotations used in the Bases References have been removed. The Writer's Guide does not require the use of quotations.

PA6 Changes have been made to be consistent with the plant specific terminology.

PA7 The title for the Bases References have been included for clarity.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 Changes have been made (additions, deletions and/or changes to the NUREG) to reflect the plant specific design references. References have been renumbered, where required.

DB2 Existing Reference 2 is actually an attachment to another document. The actual reference has been revised to reflect this other document in order to facilitate location of the references in the future.

DB3 The brackets have been removed and the proper plant specific value included.

RAI 3.1-06
RAI 3.1-07

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.6 - ROD PATTERN CONTROL

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.6

Rod Pattern Control

**RETYPED PROPOSED IMPROVED TECHNICAL
SPECIFICATIONS (ITS) AND BASES**

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

APPLICABILITY: MODES 1 and 2 with THERMAL POWER \leq 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more OPERABLE control rods not in compliance with BPWS.</p>	<p>A.1 -----NOTE----- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." ----- Move associated control rod(s) to correct position.</p>	<p>8 hours</p>
	<p><u>OR</u> A.2 Declare associated control rod(s) inoperable.</p>	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Nine or more OPERABLE control rods not in compliance with BPWS.	B.1 -----NOTE----- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1. ----- Suspend withdrawal of control rods.	Immediately
	<u>AND</u> B.2 Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6.1 Verify all OPERABLE control rods comply with BPWS.	24 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel energy deposition limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1, 6, 7, 8 and 9) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 10) and the calculated offsite doses will be well within the required limits (Ref. 5). The calculated offsite doses remain within the limits since only a small

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

number of fuel rods would reach a fuel enthalpy of 170 cal/gm, which is the enthalpy limit for eventual cladding perforation.

edit

Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel energy deposition limit will not be violated during a CRDA while following the BPWS mode of operation. The generic BPWS analysis (Ref. 11) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

The current control rod withdrawal sequence utilized at JAFNPP is known as the Reduced Notch Worth Procedure (RNWP) which was developed to reduce notch worth even further than the BPWS (Ref. 12). The CRDA analyses of References 1, 6, 7, 8, 9 and 11 bound the consequences of a CRDA for these plants following RNWP (Ref. 2).

Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 13).

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the BPWS. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

(continued)

BASES (continued)

APPLICABILITY In MODES 1 and 2, when THERMAL POWER is $\leq 10\%$ RTP, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is $> 10\%$ RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel energy deposition limit during a CRDA (Ref. 2). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

ACTIONS A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to $\leq 10\%$ RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the rod pattern, or scram if warranted.

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or reactor engineer. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram

(continued)

RAM 3.1-06

RAM 3.1-07

BASES

RAI 3.1-07

ACTIONS
(continued)

A.1 and A.2

Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or reactor engineer.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at $\leq 10\%$ RTP.

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A-13-US, General Electric Standard Application for Reactor Fuel, Supplement for United States, Sections 2.2.3.1, August 1996.
 2. Letter from T.A. Pickens (BWROG) to G.C. Laines (NRC), Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, BWROG-8644, August 15, 1986.
 3. NUREG-0979, Safety Evaluation Report Related to the Final Design Approval of the GESSAR II, BWR16 Nuclear Island Design (and Supplements 1 through 5), Section 4.2.1.3.2, April 1983.
 4. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 15.4.9, Spectrum of Rod Drop Accidents (BWR), Revision 2, July 1981.
 5. 10 CFR 100.
 6. NEDO-10527, Rod Drop Accident Analysis For Large BWRs, March 1972.
 7. NEDO-10527, Supplement 1, Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 1, Multiple Enrichment Cores With Axial Gadolinium, July 1972.
 8. NEDO-10527, Supplement 2, Rod Drop Accident Analysis For Large Boiling Water Reactors, Addendum No. 2, Exposed Cores, January 1973.
 9. NEDO-21778-A, Transient Pressure Rises Affecting Fracture Toughness Requirements For Boiling Water Reactors, December 1978.
 10. ASME, Boiler and Pressure Vessel Code, Section III, 1965 Edition, Addenda Winter of 1966.
 11. NEDO-21231, Banked Position Withdrawal Sequence, January 1977.
 12. SIL-316, Reduced Notch Worth Procedure, November 1979.
 13. 10 CFR 50.36(c)(2)(ii).
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JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

**MARKUP OF CURRENT TECHNICAL SPECIFICATIONS
(CTS)**

DISCUSSION OF CHANGES (DOCs) TO THE CTS

**NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
FOR LESS RESTRICTIVE CHANGES**

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

**JUSTIFICATION FOR DIFFERENCES (JFDs) FROM
NUREG-1433, REVISION 1**

MARKUP OF NUREG-1433, REVISION 1, BASES

**JUSTIFICATION FOR DIFFERENCES (JFDs) FROM
NUREG-1433, REVISION 1, BASES**

**RETYPE PROPOSED IMPROVED TECHNICAL
SPECIFICATIONS (ITS) AND BASES**

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

**MARKUP OF CURRENT TECHNICAL
SPECIFICATIONS (CTS)**

AI

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control System.

Objective:

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without control rods.

Specification:

A. Normal Operation

During periods when fuel is in the reactor and prior to startup from a cold condition, the Standby Liquid Control System shall be operable except as specified in 3.4.B below. This system need not be operable when the reactor is in the cold condition, all rods are fully inserted and Specification 3.3.A is met.

[LC 3.1.7]

Applicability: none / and 2

L1

[Condition A]

or can be aligned to the correct position

L4

4.4 SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The operability of the Standby Liquid Control System shall be verified by performance of the following tests:

Item	Frequency
Verify each valve (manual, power operated or automatic) in the system flowpath that is not locked, sealed or otherwise secured in position, is in the correct position.	Once per 31 Days
Pump minimum flow rate of 50 gpm shall be verified against a system head of $\geq 1,275$ psig using demineralized water from the test tank.	In accordance with the Inservice Testing Program

[SR 3.1.7.6]

MI

[SR 3.1.7.7]

LAI

add SR 3.1.7.4

MI

Specification 3.1.7

AI

JAFNPP

Verify all heat traced piping between storage tank and pump suction is unblocked

LAI

4.4 (cont'd)

LAI

SR 3.1.7.9

Manually initiate the system, except the explosive valves and pump solution in the recirculation path

Frequency
Once per 24 Months

M2

add second Frequency

SR 3.1.7.8

LAI

Explode one of three primer assemblies manufactured in same batch to verify proper function. Then install the two remaining primer assemblies of the same batch in the explosive valves.

Once per 24 Months
Staggered Test Basis

L5

SR 3.1.7.3

Demineralized water shall be injected into the reactor vessel to test that valves (except explosive valves) not checked by the recirculation test are not clogged.

Once per 24 Months

LAI

RAI 3.1-08

- 6. Test that the setting of the system pressure relief valves is between 1,400 and 1,490 psig. In accordance with the Inservice Testing Program
- 7. Disassemble and inspect one explosive valve so that it can be established that the valve is not clogged. Both valves shall be inspected within two test intervals. In accordance with the Inservice Testing Program

LAI

B. Operation with Inoperable Components

AI

From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, and continued operation permitted, provided that:

ACTION A

1. The component is returned to an operable condition within 7 days.

add ACTION B

L2

B. Operation with Inoperable Components

When a component becomes inoperable its redundant component shall be verified to be operable immediately and daily thereafter.

L3

AL

Two SLC subsystems shall be operable

JAFNPP
SR 3.1.7.10
SR 3.1.7.11
4.4 (cont'd)

3.4 (cont'd)

C. Sodium Pentaborate Solution

The standby liquid control solution tank shall contain a boron bearing solution with a minimum enrichment of 34.7 atom percent of B-10 that satisfies the volume-concentration requirements of Fig. 3.4-1 at all times when the Standby Liquid Control System is required to be operable and the solution temperature including that in the pump suction piping shall not be less than the temperature presented in Fig. 3.4-2. Tank heater and the heat tracing system shall be operable whenever the SLCS is required in order to maintain solution temperature in accordance with Fig. 3.4-2. If these requirements are not met, restore the system to the above limits within eight hours or take action in accordance with Specification 3.4.D.

LCO 3.1.7

ACTION B

ACTION C

ACTION C

C. Sodium Pentaborate Solution

The availability of the proper boron bearing solution shall be verified by performance of the following tests.

1. At least once per month - within 24 hours
Boron concentration shall be determined. In addition, the boron concentration shall be determined any time water or enriched sodium pentaborate is added or if the solution temperature drops below the limits specified by Figure 3.4-2. once it is restored within limits

2. At least once per day -

Solution volume and the solution temperature shall be checked. including pump suction piping

3. At least once per 18 months -

The temperature and level elements shall be calibrated

4. Once per 24 months -

Enrichment of B-10 (in atom percent) shall be checked

D. Not Used

D. If specifications 3.4.A through C are not met, the reactor shall be in at least hot shutdown within the following 12 hours.

SR 3.1.7.1
SR 3.1.7.2
SR 3.1.7.3
SR 3.1.7.10
SR 3.1.7.11

LA3

SR 3.1.7.5

M3

M3

A2

L31

M4

< NEW - SR 3.1.7.11

AI
↓

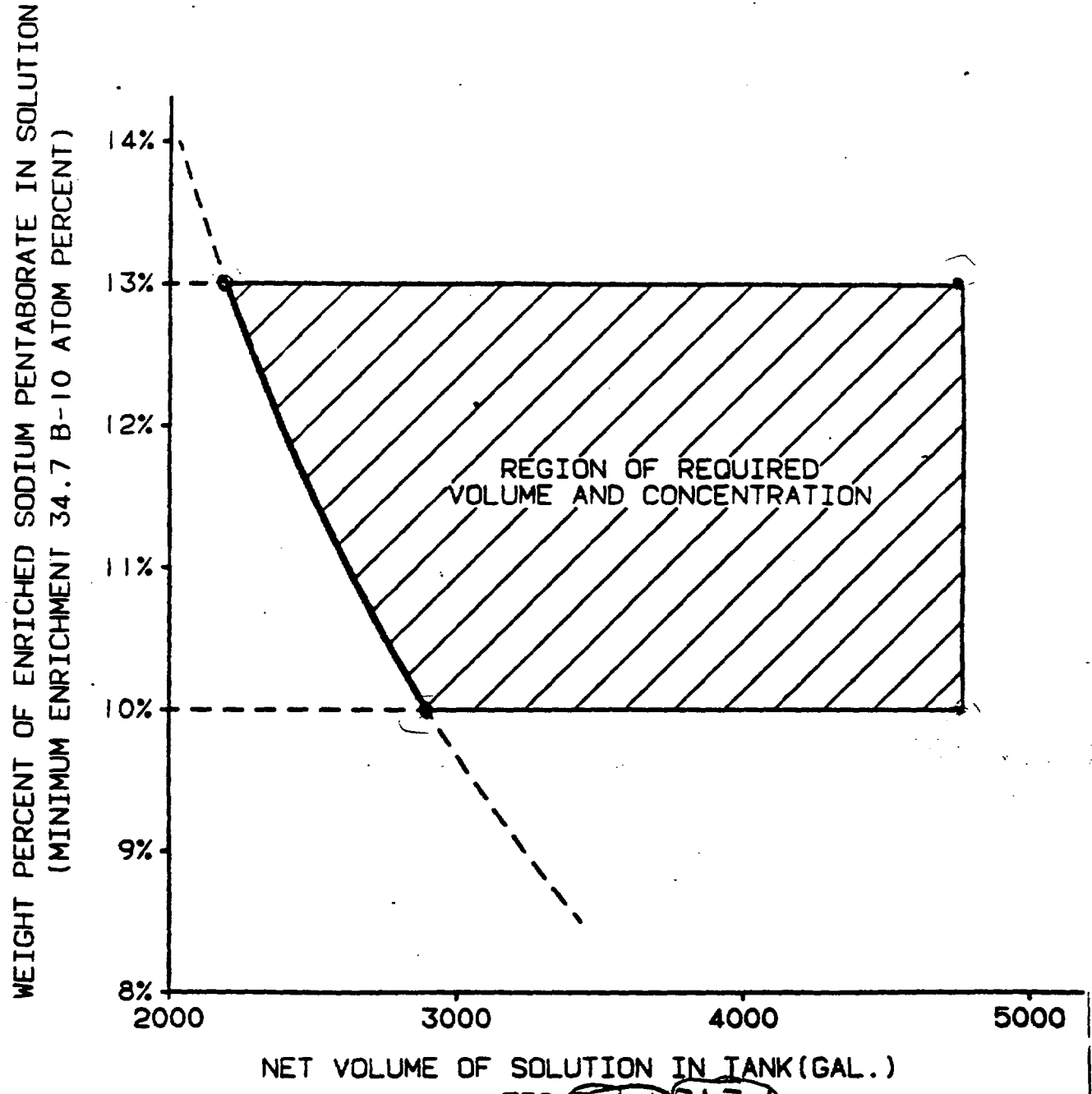


FIG. ~~3.1.7-1~~ 3.1.7-1
SODIUM PENTABORATE SOLUTION (MINIMUM 34.7 B-10 ATOM% ENRICHED)
VOLUME CONCENTRATION REQUIREMENTS.

Page 4 of 5

D-1

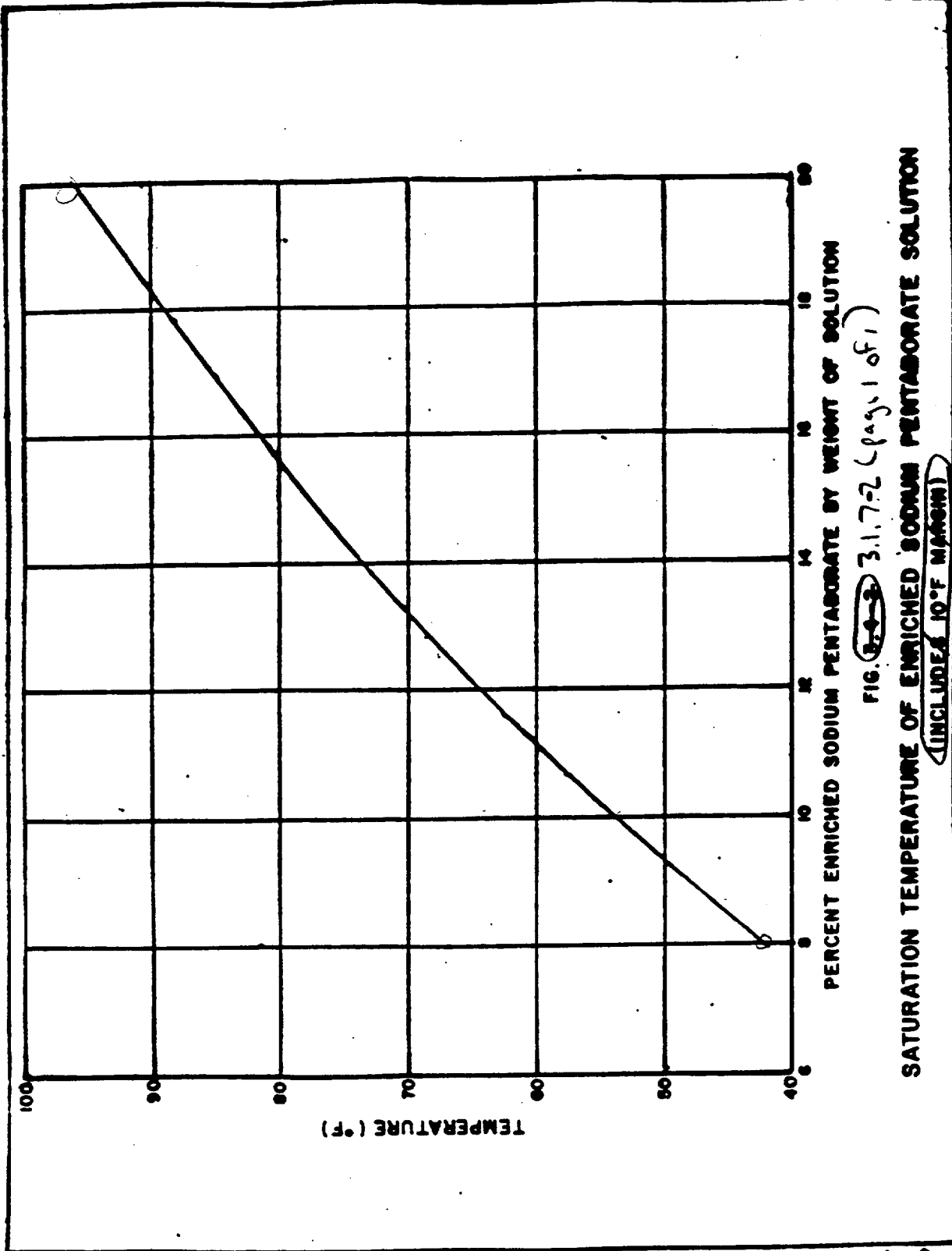


FIG. 3.1.7-2 (page 1 of 1)

3.1.7-2 (page 1 of 1)

SATURATION TEMPERATURE OF ENRICHED SODIUM PENTABORATE SOLUTION

(INCLUDES 10°F MARGIN)

L AH

111

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

**DISCUSSION OF CHANGES (DOCs) TO THE
CTS**

DISCUSSION OF CHANGES
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

ADMINISTRATIVE CHANGES

- A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).
- A2 CTS 3.4.C states that the solution temperature including the pump suction piping temperature has to be maintained above the temperature limits. CTS 4.4.C.2 requires that the solution temperature be checked at least once per day. The ITS has two separate surveillances (SR 3.1.7.2 and SR 3.1.7.3) which require that the temperature of the sodium pentaborate solution (SR 3.1.7.2) and the temperature of the pump suction piping (SR 3.1.7.3) be verified every 24 hours. Since the pump suction piping temperature has always been a requirement for SLC OPERABILITY, having a separate SR to verify this temperature is considered an administrative change. This is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - MORE RESTRICTIVE

- M1 CTS 4.4.A.1 requires the verification that all valves (manual, power operated, or automatic) in the system flowpath that is not locked, sealed or otherwise secured in position is in the correct position. There are no power operated or automatic valves in the system except for the explosive valves. This Surveillance is included as ITS SR 3.1.7.6 for all manual valves, and a new requirement has been added to verify the continuity of each explosive charge (ITS SR 3.1.7.4). Since the ITS is more explicit on the method of verification for the explosive valve this change is considered more restrictive on plant operation. This change is consistent with NUREG-1433, Revision 1.
- M2 CTS 4.4.A.3 requires verification that heat traced piping between the SLC storage tank and the pump suction is unblocked (by manually initiating the system, except explosive valves, and pump boron solution from the SLC storage tank through the recirculation path) once every 24 months. ITS SR 3.1.7.9 requires verification that heat traced piping between the SLC storage tank and the pump suction is unblocked once per 24 months and "Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2." The addition of this second Surveillance Frequency represents a more restrictive change necessary to ensure the piping is unblocked after conditions have

DISCUSSION OF CHANGES
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 (continued)

existed with the potential for causing the piping to become blocked due to precipitation of boron from solution.

M3 CTS 4.4.C.1 has requirements for checking the concentration of sodium pentaborate in the SLC Tank after certain events which could affect boron concentration occur (adding water to tank, adding boron to tank, or if temperature of solution in tank drops below the temperature limit). The CTS does not specify any time requirement for performing these checks. ITS (SR 3.1.7.5) adds a time limit of 24 hours into the requirement to check sodium pentaborate concentration after additions to the SLC Tank are made (water or boron). This ensures that the concentration is checked on a timely basis after additions to the tank are made rather than the current open ended specification. SR 3.1.7.5 also adds a second time requirement to check the concentration within 24 hours after solution temperature is restored within limits. This checks for the amount of boron that may have precipitated out of solution. The addition of new requirements reflects a more restrictive change necessary to ensure SLC System Operability is adequately maintained.

M4 CTS 4.4.C.4 requires that the enrichment of the Boron-10 (in the SLC tank) be checked once per 24 months, but the CTS contains no requirement for checking the Boron-10 enrichment of sodium pentaborate being added to the tank. ITS SR 3.1.7.10 requires that a Boron-10 enrichment verification be done prior to adding sodium pentaborate to the tank. Since the enrichment of a batch/lot of sodium pentaborate will not change with time, a single isotopic test of any given batch/lot can suffice as the required analysis for any number of mixings and additions from that batch/lot. For sodium pentaborate supplied and purchased under controls assuring appropriate 10CFR50, Appendix B and ANSI N45.2 compliance, the required analysis may be satisfied by certified vendor analytical test results. While this is consistent with current practice, this SR is considered more restrictive in that the requirement is not expressly stated in the CTS.

Once the Boron-10 is in the SLC tank the enrichment of the solution will not change. ITS SR 3.1.7.5 requires that the concentration of the boron solution in the SLC tank be verified within 24 hours after the boron addition. ITS SRs 3.1.7.1, 3.1.7.2 and 3.1.7.3 verify proper boron solution volume and temperature. ITS SR 3.1.7.11 (retained from CTS 4.4.C.4) verifies the enrichment of boron in the SLC tank every 24 months. These verifications, in addition to proposed ITS SR 3.1.7.10, help maintain the required quantity of B-10 in the tank. ITS SR 3.1.7.10 is

Revision D Change

DISCUSSION OF CHANGES
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - MORE RESTRICTIVE

M4 (continued)

considered more restrictive but provides assurance that SLC System Operability is adequately maintained. This change is consistent with NUREG-1433, Revision 1.

M5 Not used

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details of method of performing CTS 4.4.A.2 to verify flow by recirculating demineralized water to the test tank; the details in CTS 4.4.A.3, to demonstrate all piping between the SLC storage tank and the pump suction is unblocked (by manually initiating the system, except the explosive valves and pump solution in the recirculation path); the details in CTS 4.4.A.5 to verify flow through the SLC subsystem into the reactor pressure vessel (to test that the valves except explosive valves not checked by the recirculation test are not clogged); and the details in CTS 4.4.A.4 to explode one of three primer assemblies manufactured in same batch to verify proper function. (Then install the two remaining primer assemblies of the same batch in the explosive valves) are proposed to be relocated to the Bases. These details are not necessary to ensure that the SLC System is maintained Operable. The requirements of ITS 3.1.7 and SRs 3.1.7.7, 3.1.7.8, and 3.1.7.9 are adequate to ensure the capability to provide flow through each SLC subsystem to the test tank and into the reactor pressure vessel, to ensure the piping between the SLC storage tank and the pump suction is unblocked, and to ensure SLC System Operability. Therefore, the relocated details are not necessary to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA2 The testing requirements of CTS 4.4.A.6 (to verify the proper operation and setpoints of the relief valves) and CTS 4.4.A.7 (to disassemble and inspect one explosive valve) are proposed to be relocated to the Inservice Testing (IST) Program. These testing requirements demonstrate the SLC System relief valves and explosive valves are OPERABLE. However, the IST Program, required by 10 CFR 50.55a, provides requirements for the testing of all ASME Code Class 1, 2, and 3 valves in accordance with Section XI of the ASME Code. ITS Section 5.5.7 provides controls over the IST Program. These controls are adequate to ensure the required testing to demonstrate Operability is performed. Therefore, the relocated requirements are not necessary to be in the ITS

RAI REV D
3.1.08 CHANGE

DISCUSSION OF CHANGES
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC) (continued)

LA2 (continued)

to provide adequate protection of the public health and safety. Changes to the relocated requirements in the IST Program will be controlled by the provisions of 10 CFR 50.59.

- LA3 CTS 3.4.C contains detailed information concerning the boron solution for the SLC storage tank, and what support components and variables are required to assure SLC OPERABILITY is maintained. The ITS relocates this detailed information to the Bases for Specification 3.1.7. The requirements of ITS 3.1.7 including the LCO, ACTIONS and Surveillances are adequate to ensure SLC System OPERABILITY. Therefore, the relocated details are not necessary to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.
- LA4 The detail in CTS Figure 3.4-2 that the saturation temperature of enriched sodium pentaborate solution curve includes a 10°F margin is proposed to be relocated to the Bases. The requirements in ITS SR 3.1.7.2 to verify the temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2 (CTS Figure 3.4-2) and Figure 3.1.7-2 (Sodium Pentaborate Solution Temperature Versus Concentration Requirements curve) are adequate to ensure the proper evaluation is performed and therefore help ensure SLC System OPERABILITY. Therefore, the relocated details are not necessary to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.
- LB1 The requirements in CTS 4.4.C.3 to calibrate the temperature and level elements is proposed to be relocated to the Technical Requirements Manual (TRM). These temperature and level indications do not necessarily relate directly to SLC System OPERABILITY. In general NUREG-1433, Revision 1, does not specify requirements for equipment which only provide indication to support OPERABILITY of a system or component. Control of the availability of, and necessary compensatory activities if not available, for indications, monitoring instruments, and alarms are addressed by plant operational procedures and policies. Therefore, the SLCs temperature and level instrument surveillances are removed from the Technical Specifications and relocated to the TRM. The requirements in ITS 3.1.7 including the LCO, ACTIONS and Surveillances are adequate to ensure the SLC System is Operable. Therefore, the relocated requirements are not necessary to be in the ITS to provide adequate protection of the public health and safety. At ITS

DISCUSSION OF CHANGES
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC) (continued)

LBI (continued)

implementation, the relocated requirement will be incorporated by reference into the UFSAR. As such changes to the relocated requirements in the TRM will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

- L1 CTS requires that the Standby Liquid Control System be Operable during a period when fuel is in the reactor and prior to startup from cold condition. This System need not be Operable when the reactor is in the cold condition, control rods are fully inserted and CTS 3.3.A (Reactivity Limitations) is met. The ITS 3.1.7 Applicability is MODES 1 and 2. The current Applicability corresponds to MODES 1, 2 and may even imply MODES 3, 4 and 5 with any control rod withdrawn. This change is less restrictive since the new Applicability does not include MODES 3, 4, and 5. The SLC system is not needed during Hot or Cold Shutdown (MODES 3 or 4) since control rods can only be withdrawn in accordance with Section 3.10, "Special Operations," and adequate SDM prevents criticality under these conditions. While in the refueling MODE, the SLC System is not needed because only a single control rod can be withdrawn and adequate SDM prevents criticality when under these conditions.
- L2 CTS 3.4.C includes an action to restore certain components (e.g., tank heaters) or variables (e.g., sodium pentaborate volume-concentration and temperature requirements) within 8 hours or take action to be in hot shutdown in the next 12 hours. All the components or variables discussed in CTS 3.4.C will cause both subsystems of the SLC System to be inoperable. However, the list is not all inclusive of the possible events which could lead to both subsystems being inoperable. ITS 3.1.7, ACTION B is being added to allow the entire SLC System (e.g., both pumps) to be inoperable for any reason up to 8 hours prior to requiring a plant shutdown. The 8 hours provides time to restore minor problems (e.g., some pump inoperabilities) prior to requiring a plant shutdown. The 8 hours is considered acceptable since the time is short, the SLC System is not the primary method of shutting down the plant, reduces the possibility of plant shutdown transients and the probability of an ATWS event is very small.

DISCUSSION OF CHANGES
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

- L3 CTS 4.4.B requires that when a SLC subsystem or component becomes inoperable, the redundant subsystem or component be verified to be OPERABLE immediately and daily thereafter. ITS 3.1.7 does not have this cross system check. This change will allow credit to be taken for normal periodic Surveillances as a verification of OPERABILITY and availability of the remaining SLC subsystem. The periodic Frequencies specified to verify OPERABILITY of the remaining SLC subsystem has been shown to be adequate to ensure equipment OPERABILITY. As stated in NRC Generic Letter 87-09, "It is overly conservative to assume that systems or components are inoperable when a surveillance requirement has not been performed. The opposite is in fact the case; the vast majority of surveillances demonstrate the systems or components in fact are operable." Therefore, reliance on the specified Surveillance intervals does not result in a reduced level of confidence concerning the equipment availability. The ITS and current BWR operating philosophy accept the philosophy of system OPERABILITY based on satisfactory performance of monthly, quarterly, refueling interval, post-maintenance or other specified performance tests without requiring additional testing when another system is inoperable (except for diesel generator testing, which is not being changed).
- L4 CTS 4.4.A.1 requires that each SLC subsystem "valve (manual, power operated, or automatic) in the system flow path that is not locked, sealed or otherwise secured in position, is in the correct position" once per 31 days. ITS SR 3.1.7.6 requires that "each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position" every 31 days. The proposed change permits the SLC subsystem to be considered OPERABLE as long as the valves can be manually realigned to their correct position. The Bases stipulates that this realignment must be capable of being done from the control room, or locally by a dedicated operator at the valve control. The SLC System is a manually initiated system. Therefore allowing the system to be considered OPERABLE whenever the system valves can be correctly aligned does not reduce the level of safety and is considered acceptable. This change is consistent with NUREG-1433, Revision 1.

DISCUSSION OF CHANGES
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

- L5 CTS 4.4.A.5 requires that every 24 months demineralized water be injected into the reactor vessel to test that valves (except explosive valves) not checked by the recirculation test (CTS 4.4.A.3) are not clogged. This test involves testing entire subsystems; including portions common to both subsystems as well as non-common portions. As such, testing either subsystem can satisfy the necessary testing for the common portions of both subsystems. To accomplish this, ITS SR 3.1.7.8 requires the verification of flow through one SLC subsystem from the pump into reactor pressure vessel every 24 months on a STAGGERED TEST BASIS (i.e., such that the subsystems use for the test are alternated each 24 months). Since the CTS could be inferred to require testing both subsystems each 24 months, this change is a relaxation in the frequency of testing an individual subsystem (i.e., on the Staggered Test Basis), and is classified as a less restrictive change. Testing of the non-common portions, which are also the subject of the relaxed testing frequency, are appropriately surveilled by other ITS SRs: specifically, each pump is tested per the IST Program as required by SR 3.1.7.7, the continuity of each explosive charge is verified every 31 days in accordance with SR 3.1.7.4, the temperature of pump suction piping is verified within limits every 24 hours with SR 3.1.7.3, and proper manual valve position is verified every 31 days as required by SR 3.1.7.6. These surveillance tests, which are performed more frequently than the proposed surveillance interval of SR 3.1.7.8, provide assurance that unacceptable conditions associated with the SLC System will be detected in a timely manner. This change is also consistent with NUREG-1433, Revision 1.

RAI 3.1-08

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

**NO SIGNIFICANT HAZARDS CONSIDERATION
(NSHC) FOR LESS RESTRICTIVE CHANGES**

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. 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?

The proposed change deletes the requirements for Standby Liquid Control (SLC) System operability during Hot Shutdown, Cold Shutdown and Refueling with any control rod withdrawn from the core. The SLC System is not assumed in the initiation of any previously evaluated events and therefore the proposed change will not increase the probability or consequence of a previously analyzed accident. The SLC System is not assumed to operate in the mitigation of any previously analyzed accidents which are assumed to occur during Hot Shutdown, Cold Shutdown or Refueling. This change will not result in operation that will increase the probability of initiating an analyzed event. This change will not alter assumptions relative to mitigation of an accident or alter the operation of process variables, structures, systems, or components as described in the safety analyses. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

This proposed change relaxes the modes of applicability for the SLC Specification. The proposed change will not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose or eliminate any requirements. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change would remove a backup (in MODES 3, 4 and 5) to the available systems for reactivity control. However, this backup is not considered in the margin of safety when determining the required

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

3. (continued)

reactivity for shutdown and refueling events. This change will have no impact on any safety analysis assumptions since the SLC System will be required to be Operable in Modes 1 and 2. As such, no question of safety is involved. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. 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?

The proposed change allows 8 hours to restore one SLC subsystem to operable status when both subsystems are inoperable for any reason. The SLC System is not identified as an initiator for any accidents previously analyzed, and therefore, this expanded coverage of inoperable components or processes which make the SLC System inoperable, will not significantly increase the probability of an accident previously evaluated. The probability of an ATWS accident occurring while the SLC System is inoperable is very small. The SLC System provides backup protection only in case the control rods do not shutdown the reactor. The consequences of an ATWS accident during this 8 hour period when both SLC subsystems are inoperable for other reasons will be bounded by the current allowance of 8 hours for limited inoperabilities in CTS 3.4.C (e.g., volume-concentration, temperature). In addition, this change provides the benefit of potentially avoiding a plant shutdown transient (due to the 8 hour Completion Time) when both SLC subsystems are inoperable. Therefore, this change will not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change allows 8 hours to restore one SLC subsystem to operable status when both subsystems are inoperable for any reason. This change will have no impact on any safety analysis assumptions. As such, no question of safety is involved. The SLC system provides backup protection only in case the control rods do not shutdown the

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

3. (continued)

reactor. The consequences of an ATWS accident during this 8 hour period when both SLC subsystems are inoperable for other reasons will be bounded by the current allowance of 8 hours for limited inoperabilities in CTS 3.4.C (e.g., volume-concentration, temperature). In addition, this change provides the benefit of potentially avoiding a plant shutdown transient (due to the 8 hour Completion Time) when both SLC subsystems are inoperable. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. 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 does not result in any hardware or operating procedure changes. The SLC System is not assumed to be an initiator of any analyzed event. This change redefines the method for verifying Operability of the remaining subsystem when a subsystem is declared inoperable. The periodic frequencies specified to demonstrate Operability of the remaining components have been shown to be adequate to ensure equipment Operability. Since the other subsystem remains Operable, redefining the method by which the subsystem is verified Operable does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will only redefine the method by which the remaining subsystem is verified Operable when the other is declared inoperable. Redefining the method by which a subsystem is verified operable does 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 allows credit to be taken for normal periodic surveillances as a demonstration of Operability and availability of the remaining SLC subsystem. Thus, this change eliminates the requirement to perform surveillances on a subsystem when the other is declared inoperable. The periodic frequencies specified to demonstrate Operability of the remaining components have been shown to be adequate to ensure equipment Operability. As stated in NRC Generic Letter 87-09, "It is overly conservative to assume that systems or components are inoperable when a

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

3. (continued)

surveillance requirement has not been performed. The opposite is in fact the case; the vast majority of surveillances demonstrate the systems or components in fact are operable." Therefore, reliance on the specified surveillance intervals does not result in a reduced level of confidence concerning the equipment availability. Reliance on the normal surveillance requirement is judged to be an equivalent testing program as compared to the requirements being deleted. Thus, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

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

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

CTS 4.4.A.1 requires that each SLC subsystem "valve (manual, power operated, or automatic) in the system flow path that is not locked, sealed or otherwise secured in position, is in the correct position" once per 31 days. ITS SR 3.1.7.6 requires that "each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position" every 31 days. The proposed change permits the SLC subsystem to be considered Operable as long as the valves can be manually realigned to their correct position. The Bases stipulates that this realignment must be capable of being done from the control room, or locally by a dedicated operator at the valve control. The SLC System is a manually initiated system. As such it is not the initiator of any accident previously evaluated. Therefore, the probability of any previously evaluated accident can not increase. The proposed change does not change the system capability or any assumed response time (since it is a manually initiated system). Therefore, the consequences of any previously evaluated accident has not changed.

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

The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does 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 will have no impact on any safety analysis assumptions. As such, no question of safety is involved. The SLC system provides manual backup scram protection only in case the control rods do not shutdown the reactor. Permitting the system to be manually aligned does not change the ability of the system to perform its intended function.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

3. (continued)

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. 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?

The proposed change involves a change in the surveillance testing intervals for a portion of each subsystem from 24 to 48 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the SRs themselves nor the way in which the surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because other tests performed more frequently will identify potential equipment problems. Furthermore, a historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

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

The proposed change does not introduce any failure mechanism of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the SRs themselves and the way surveillances are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

RAM 31-08

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

RAM 3.1-08

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

Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is small based on other, more frequent testing, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

**MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION**

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

[3.4.A] LCO 3.1.7 Two SLC subsystems shall be OPERABLE.
[3.4.C]

[L1] APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of boron in solution not within limits but > [].	A.1 Restore concentration of boron in solution to within limits.	72 hours AND 10 days from discovery of failure to meet the LCO
[3.4.B.1] A.B. One SLC subsystem inoperable [for reasons other than Condition A].	[A] B.1 Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
[L2] [3.4.C] B.C. Two SLC subsystems inoperable [for reasons other than Condition A].	[B] C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
[3.4.D] C.D. Required Action and associated Completion Time not met.	[C] D.1 Be in MODE 3.	12 hours





(DBI)

BWR/4 STS JAFNPP

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Typ All Pages

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>[4.4.C.2] SR 3.1.7.1 Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1. (or 2 - 4536) gallons. DB2</p>	<p>24 hours DB2</p>
<p>[4.4.C.2]  SR 3.1.7.2 Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2. DB2</p>	<p>24 hours  DB2</p>
<p>[4.4.C.2]  SR 3.1.7.3 Verify temperature of pump suction piping is within the limits of Figure 3.1.7-2. DB2</p>	<p>24 hours  DB2</p>
<p>[M1] SR 3.1.7.4 Verify continuity of explosive charge.</p>	<p>31 days</p>
<p>[4.4.C.1] [M3] SR 3.1.7.5 Verify the concentration of boron in solution is within the limits of Figure 3.1.7-1. DB2</p>	<p>31 days <u>AND</u> Once within 24 hours after water or boron is added to solution <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2. DB2</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.1.7.6 Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.</p> <p>[4.4.A.1] [M1] [L4] DB3</p>	<p>31 days</p>
<p>SR 3.1.7.7 Verify each pump develops a flow rate \geq [41.2] gpm at a discharge pressure \geq [190] psig.</p> <p>[4.4.A.2] 50 1275 DB2</p>	<p>In accordance with the Inservice Testing Program of 32 days</p> <p>CLB1</p>
<p>SR 3.1.7.8 Verify flow through one SLC subsystem from pump into reactor pressure vessel.</p> <p>[4.4.A.4] [4.4.A.5] [M5]</p>	<p>²⁴ 18 months on a STAGGERED TEST BASIS CLB2</p>
<p>SR 3.1.7.9 Verify all heat traced piping between storage tank and pump suction is unblocked.</p> <p>[4.4.A.3] [M2] CLB2</p>	<p>²⁴ 18 months AND Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2) CLB2</p>
<p>SR 3.1.7.10 Verify sodium pentaborate enrichment is \geq [60.0] atom percent B-10.</p> <p>[4.4.C.4] [M4] 34.7 DB2</p>	<p>Prior to addition to SLC tank DB5</p>

[4.4.C.4] BWR/4 STS INSERT SR 3.1.7.11 CLB3

3.1-22

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NEW - SR 3.1.7.11

CLB 3

INSERT SR 3.1.7.11

SR 3.1.7.11 Verify sodium pentaborate enrichment in solution in the SLC tank is ≥ 34.7 atom percent B-10.

24 months

<NEW>

DB4

INSERT
1

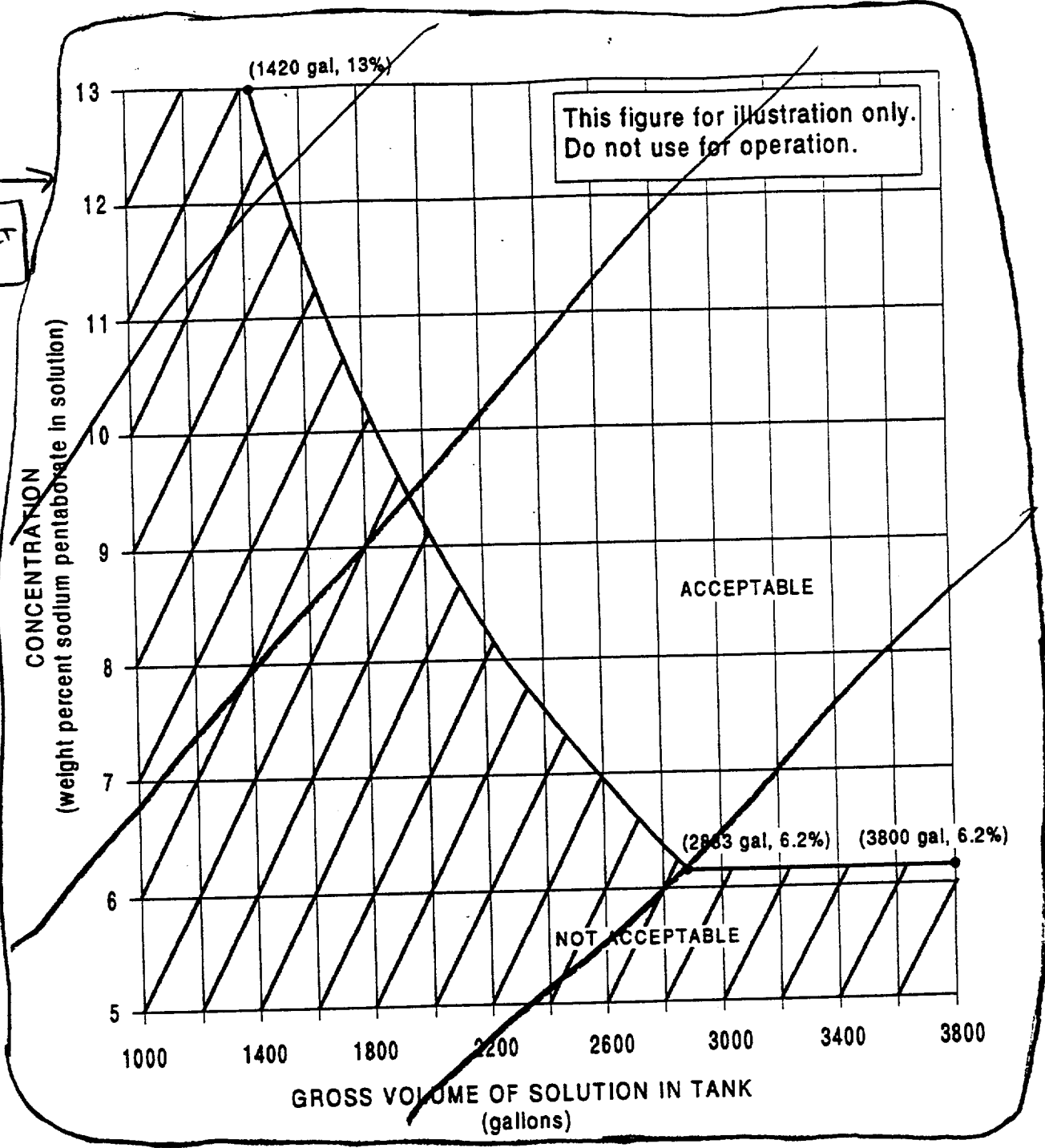
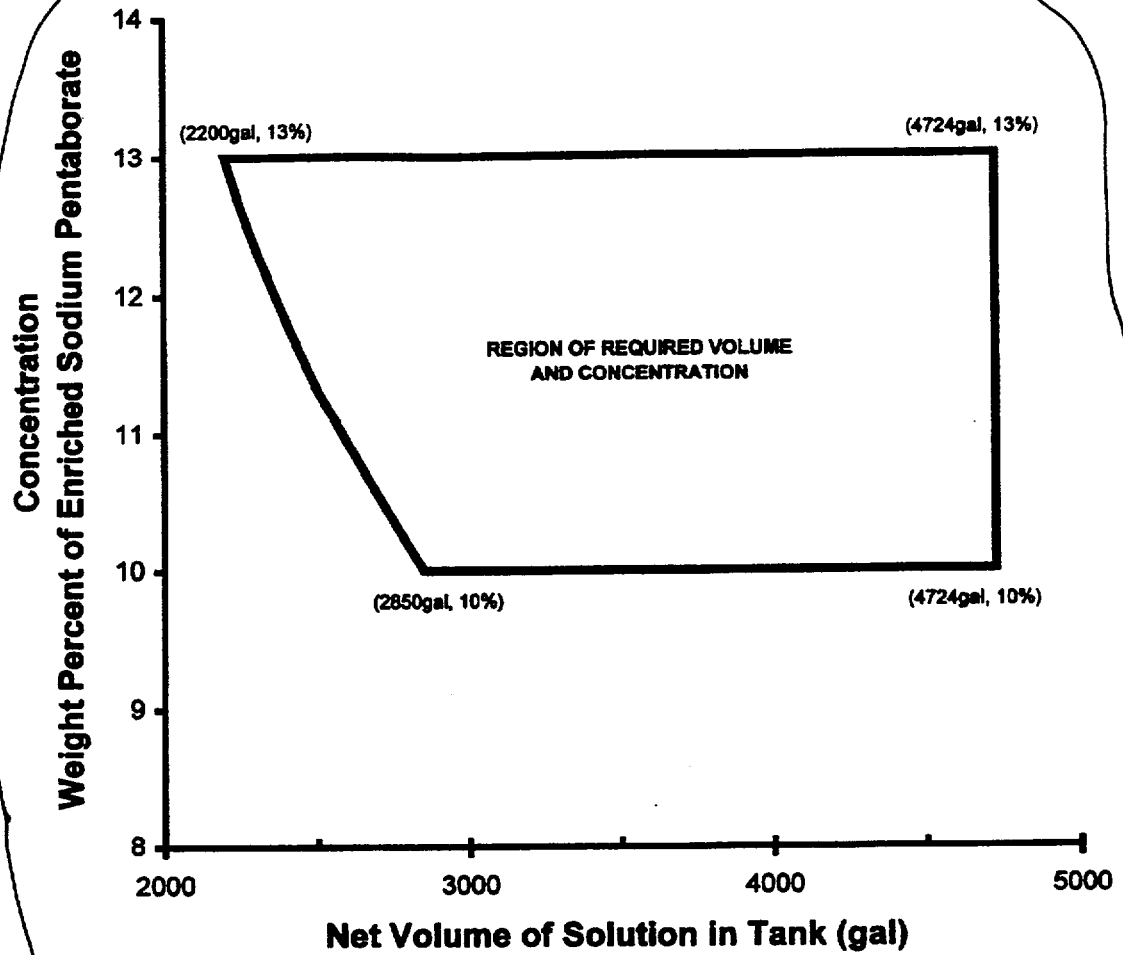


Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

INSERT 1



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Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

DB4

INSECT
2

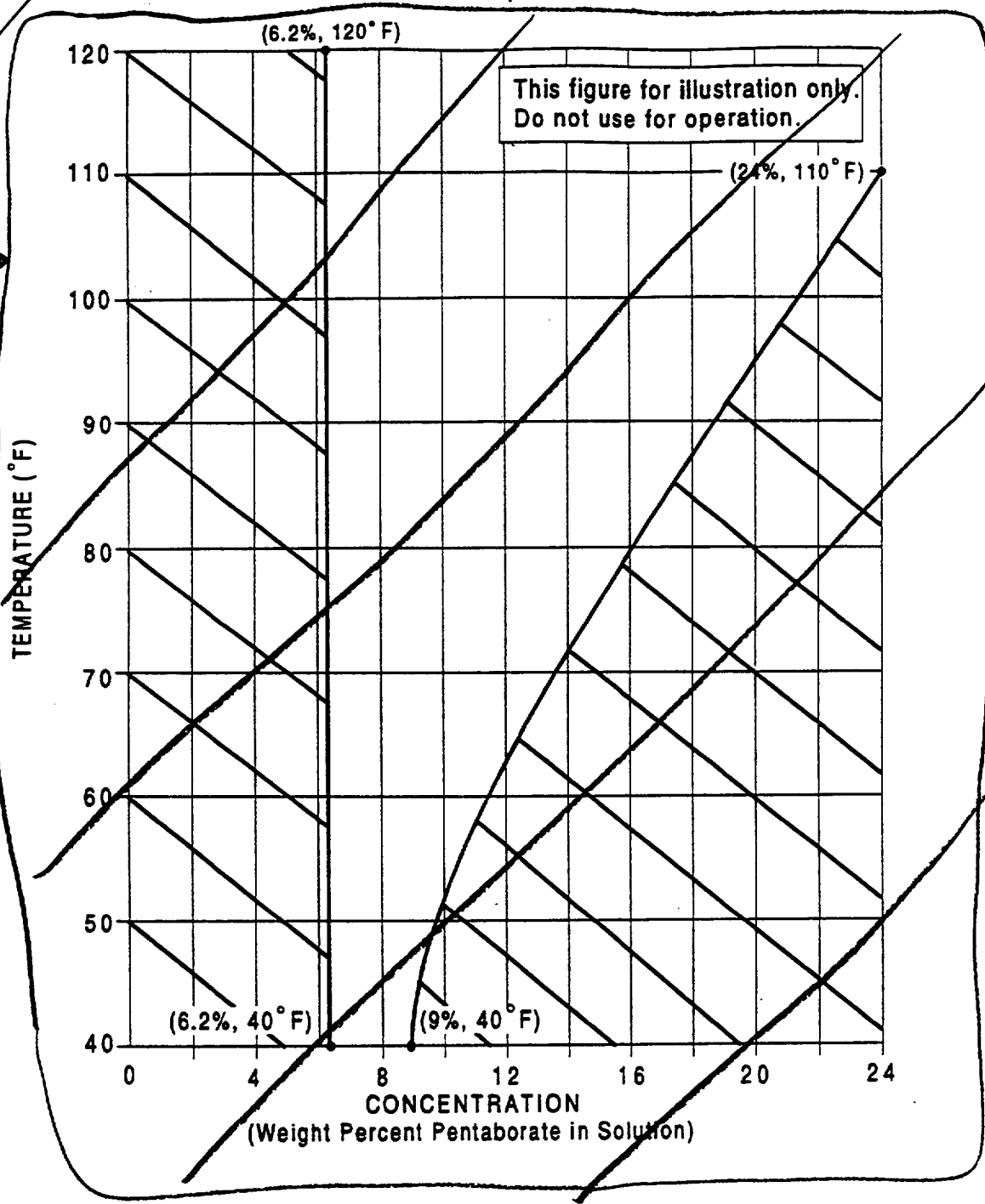
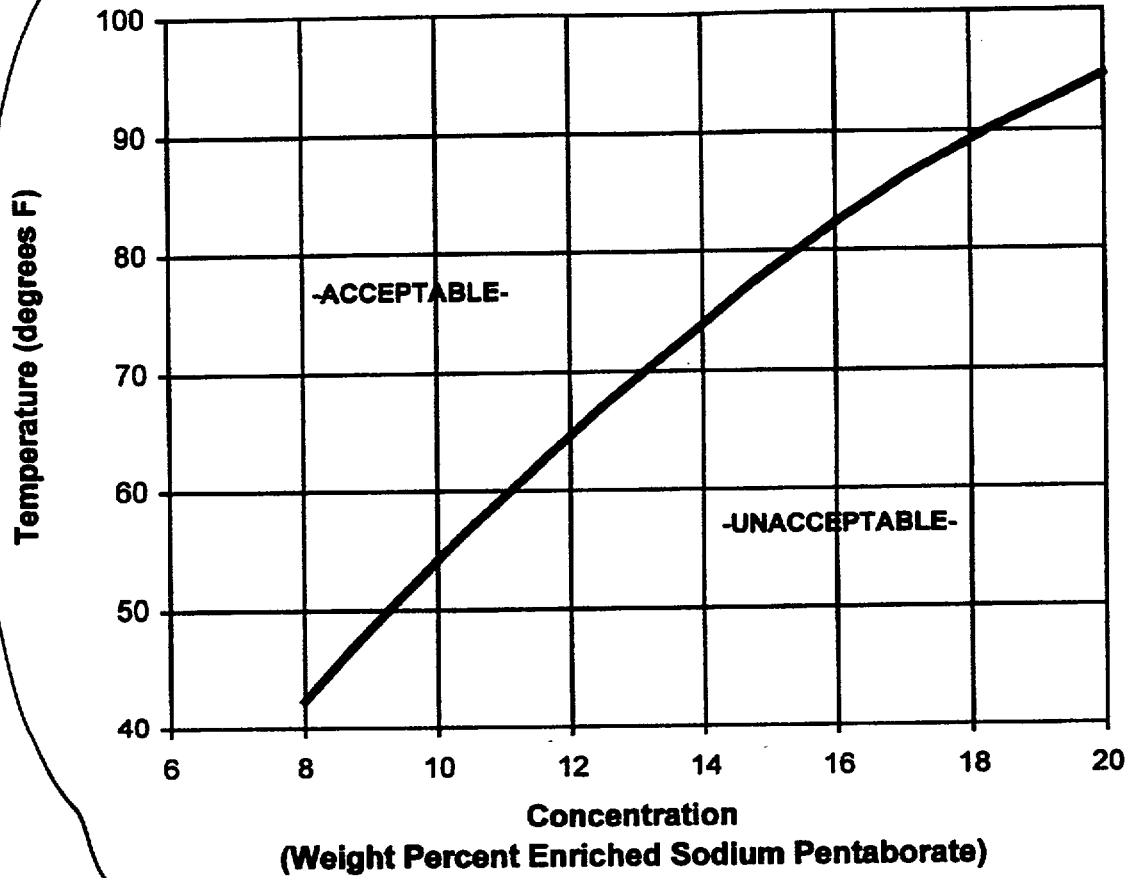


Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

INSERT 2



D

Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution
Temperature Versus Concentration Requirements

JAFNPP

INSERT pg. 3.1-24

Amendment

REVISION D

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

**JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1**

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

- CLB1 The brackets have been removed and the Frequency "In accordance with the Inservice Testing Program" retained consistent with the current licensing basis in CTS 4.4.A.2.
- CLB2 The brackets have been removed and a 24 month Frequency included in SR 3.1.7.8 and SR 3.1.7.9 (first frequency) consistent with CTS 4.4.A.4 and CTS 4.4.A.3, respectively. In addition, the brackets have been removed from SR 3.1.7.9 and the second frequency has been added in accordance with M2.
- CLB3 The requirements of CTS 4.4.C.4 for verifying Boron-10 enrichment of the sodium pentaborate solution in the SLC tank on a 24 month Frequency are retained as SR 3.1.7.11. Although the provisions of SR 3.1.7.10 are adequate to ensure proper Boron-10 enrichment, periodic verification of the SLC solution enrichment is a good practice providing added assurance that the proper Boron-10 enrichment is maintained.



PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

- DB1 NUREG-1433 ACTION A, is not applicable to JAFNPP and, has been deleted. JAFNPP requires the same concentration of boron in solution to meet the original licensing basis of the SLC System (cold shutdown) as it does for the ATWS rule (10 CFR 50.62). Therefore low boron concentration would result in both SLC subsystems being inoperable. The remaining Conditions and Required Actions have been renumbered or revised to reflect this deletion.
- DB2 The brackets have been removed and the proper limits included.
- DB3 The brackets have been removed and the information deleted since the system does not include any power operated or automatic valves other than the explosive valves.
- DB4 NUREG Figures 3.1.7-1 and Figure 3.1.7-2 have been modified in accordance with the current requirements.
- DB5 The brackets have been removed and SR 3.1.7.10 retained in accordance with CTS 4.4.C.4 and DOC M4.



JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

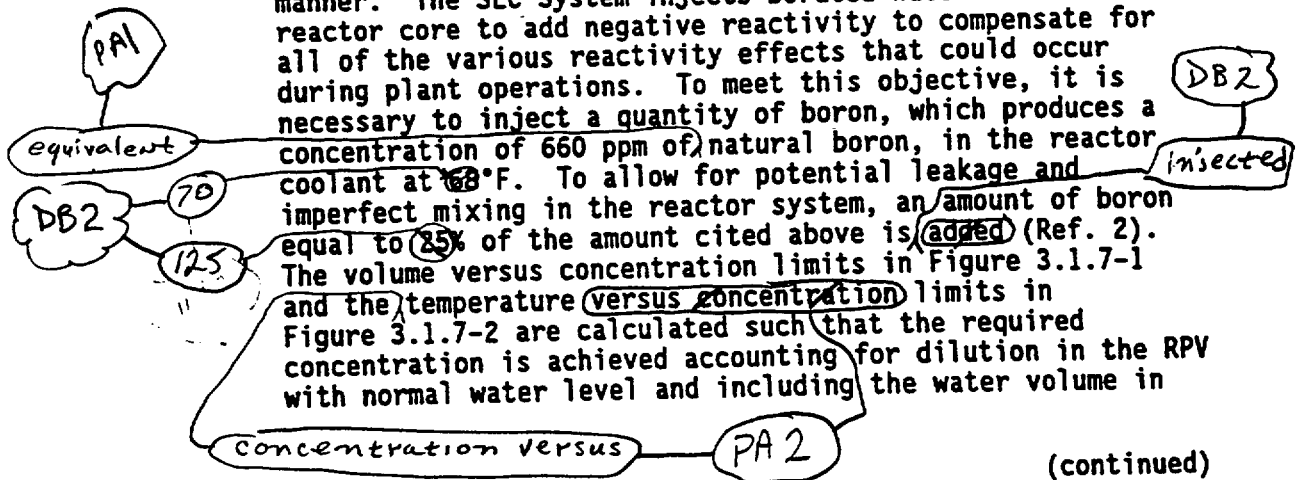
BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of natural boron, in the reactor coolant at 68°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the temperature versus concentration limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in



(continued)

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B 3.1-39

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REVISION D

BASES

(6 inches above tank bottom)

DB2

APPLICABLE
SAFETY ANALYSES
(continued)

the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction ~~shutdown~~ level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

Criterion 4 of
10 CFR 50.36 (2)(2)(ii)
(Ref. 3)

TA-1 & XI

The SLC System satisfies the requirements of the NRC Policy Statement because operating experience and probabilistic risk assessments have shown the SLC System to be important to public health and safety. Thus, it is retained in the Technical Specifications.

TSTP-367

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

If the boron solution concentration is less than the required limits for mitigation but greater than the

DB1

(continued)

BASES

ACTIONS

A.1 (continued)

concentration required for cold shutdown (original licensing basis), the concentration must be restored to within limits in 72 hours. It is not necessary under these conditions to enter Condition C for both SLC subsystems inoperable since they are capable of performing their original design basis function. Because of the low probability of an event and the fact that the SLC System capability still exists for vessel injection under these conditions, the allowed Completion Time of 72 hours is acceptable and provides adequate time to restore concentration to within limits.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, an SLC subsystem is inoperable and that subsystem is subsequently returned the OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total duration of 10 days (7 days in Condition B, followed by 3 days in Condition A), since initial failure of the LCO, to restore the SLC System. Then an SLC subsystem could be found inoperable again, and concentration could be restored to within limits. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition A was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

If one SLC subsystem is inoperable for reasons other than Condition A, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown

(continued)

BASES

ACTIONS

A.1.1 (continued)

capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the Plant reactor.

Control rods

PAZ

DBI

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of concentration out of limits or inoperable SLC subsystems during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, concentration is out of limits, and is subsequently returned to within limits, the LCO may already have been not met for up to 3 days. This situation could lead to a total duration of 10 days (3 days in Condition A, followed by 7 days in Condition B), since initial failure of the LCO, to restore the SLC System. Then concentration could be found out of limits again, and the SLC subsystem could be restored to OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock," resulting in establishing the "time zero" at the time the LCO was initially not met instead of at the time Condition B was entered. The 10 day Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

DBI

B.1.1

DBI

If both SLC subsystems are inoperable for reasons other than Condition A, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or severe transient occurring concurrent with the failure of the control rods to shut down the reactor.

PAZ

(continued)

BASES

DBI

ACTIONS
(continued)

3.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction piping. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained above the saturation temperature. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for

(continued)

BASES

SURVEILLANCE
REQUIREMENT

SR 3.1.7.4 and SR 3.1.7.6 (continued)

DB4

manual, power operated, and automatic valves in the SLC System flow path provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance also 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 verification of valve alignment 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 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

PA2

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. SR 3.1.7.5 must also be performed anytime the temperature is restored to within the limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

is maintained per Figure 3.1.7-1
PA2

PA2

PA3

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 2195 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration

DB2

1275

by ferric lab by demineralized water to the test tank

50
DB2

surveillance interval.
(continued)

PA2

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.7 (continued)

PA2
pump and motor
capability

requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is [in accordance with the Inservice Testing Program or 92 days].

PA2

tests

SR 3.1.7.8 and SR 3.1.7.9

primer
assembly

CLB1

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 36 months at alternating 18 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 36 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 36 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Primer
assembly

CLB2

48

CLB2

pathway

CLB2

24

Upon completion of this verification, the pump section piping must be flushed with demineralized water to ensure piping between the storage tank and pump section is unblocked.

PA3

manually initiate the system, except the explosive valves, and

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.8 and SR 3.1.7.9 (continued)

24
CLB2

The 12 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored to within the limits of Figure 3.1.7-2.

SR 3.1.7.10

PA4

INSERT
SR3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used.

See JFD PA4

REFERENCES

1. 10 CFR 50.62.
2. WFSAR, Section ~~[4.2.3.4.5]~~ 3.9.4

PA1

DB3

3. 10 CFR 50.36 (c)(2)(ii)

INSERT
SR 3.1.7.11

CLB3

X1

See SFD CLB3
NEW

PA4

INSERT for SR 3.1.7.10

A single isotopic test from a single batch can suffice the required analysis for any number of mixings and additions from this batch. Certified vendor analytical test results may be used to satisfy this requirement.

△

CLB3

INSERT for SR 3.1.7.11

The B-10 enrichment of boron in solution in the SLC tank is only affected by the B-10 enrichment of tank additions. The requirements of SR 3.1.7.10 serve to assure that tank additions contain the proper enrichment. SR 3.1.7.11 requires periodic verification of the B-10 enrichment of the solution in the SLC tank, providing added assurance that the proper B-10 enrichment is maintained.

MEMO

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

**JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1, BASES**

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

RETENTION OF EXISTING REQUIREMENT (CLB)

- CLB1 The frequency in SR 3.1.7.7 has been retained in accordance with CTS 4.4.A.2.
- CLB2 The Frequency in SR 3.1.7.8 and SR 3.1.7.9 have been retained in accordance with CTS 4.4.A.4 and 4.4.A.3, respectively. The wording has been revised to retain the current method of testing the primer assemblies. In addition 36 months has been increased to 48 months consistent with the current 24 month Frequency.
- CLB3 The requirements of CTS 4.4.C.4 for verifying Boron-10 enrichment of the sodium pentaborate solution in the SLC tank on a 24 month Frequency are retained as SR 3.1.7.11. Although the provisions of SR 3.1.7.10 are adequate to ensure proper Boron-10 enrichment, periodic verification of the SLC solution enrichment is a good practice providing added assurance that the proper Boron-10 enrichment is maintained.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

- PA1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.
- PA2 Editorial change made for enhanced clarity or to be consistent with similar statements made in other places in the Specifications or Bases.
- PA3 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific method to perform the Surveillance.
- PA4 Clarification of the intent of the B-10 enrichment verification is added. Since enrichment will not vary over time, once verification is completed for any single batch of granular sodium pentaborate, it remains valid for all additions from that batch.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

- DB1 NUREG-1433 ACTION A, is not applicable to JAFNPP and, has been deleted. JAFNPP requires the same concentration of boron in solution to meet the original licensing basis of the SLC System (cold shutdown) as it does for the ATWS rule (10 CFR 50.62). Therefore low boron concentration would result in both SLC subsystems being inoperable. The remaining Conditions and Required Actions have been renumbered or revised to reflect this deletion.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.7 - STANDBY LIQUID CONTROL (SLC) SYSTEM

- DB2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific analysis.
- DB3 The brackets have been removed from the References and the appropriate JAFNPP reference included.
- DB4 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific design.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

- TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler number 367, Revision 0, have been incorporated into the revised Improved Technical Specifications.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

- X1 NUREG-1433, Revision 1, Bases references to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

TSTF-367

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.7

Standby Liquid Control (SLC) System

**RETYPE PROPOSED IMPROVED TECHNICAL
SPECIFICATIONS (ITS) AND BASES**

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

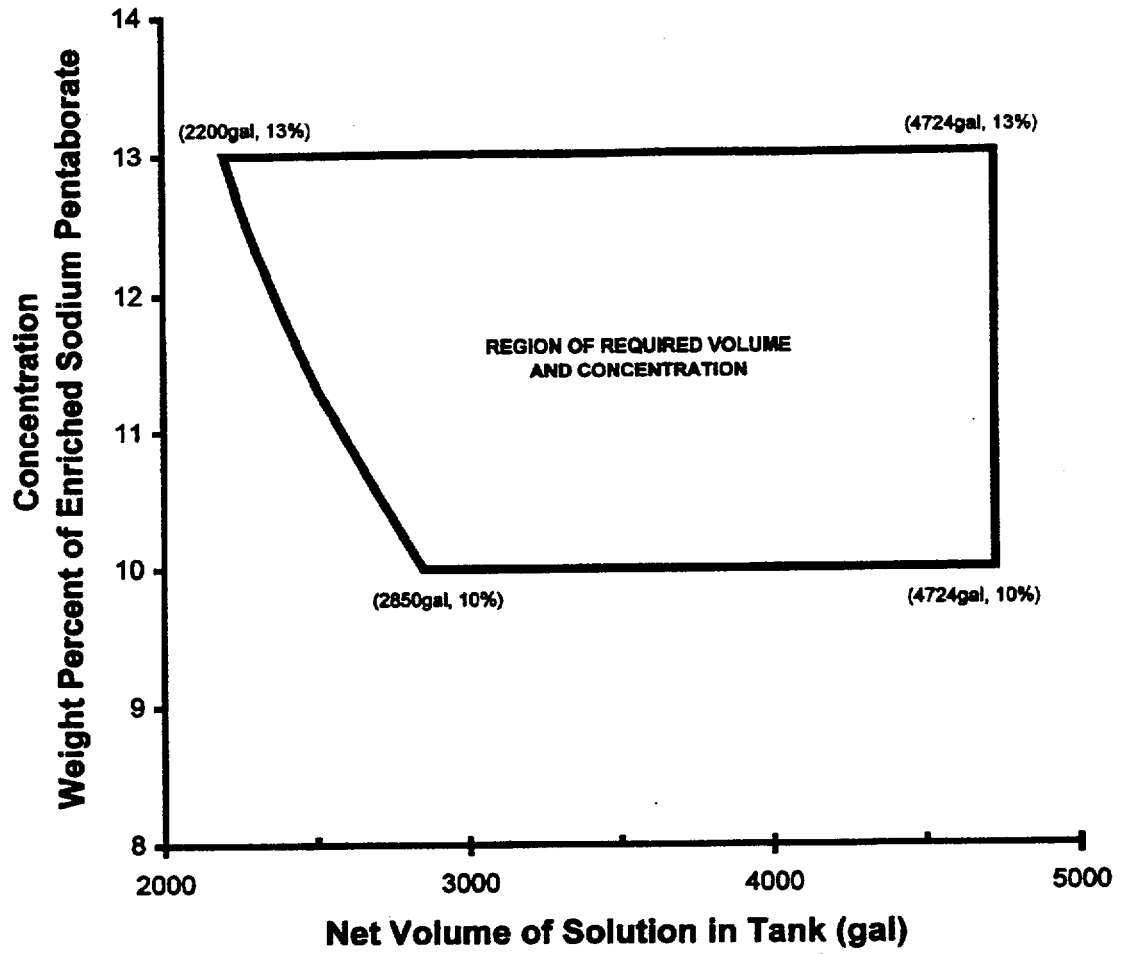
SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify available volume of sodium pentaborate solution is within the limits of Figure 3.1.7-1.	24 hours
SR 3.1.7.2 Verify temperature of sodium pentaborate solution is within the limits of Figure 3.1.7-2.	24 hours
SR 3.1.7.3 Verify temperature of pump suction piping is within the limits of Figure 3.1.7-2.	24 hours
SR 3.1.7.4 Verify continuity of explosive charge.	31 days
SR 3.1.7.5 Verify the concentration of boron in solution is within the limits of Figure 3.1.7-1.	31 days <u>AND</u> Once within 24 hours after water or boron is added to solution <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2

(continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 50 gpm at a discharge pressure ≥ 1275 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	24 months AND Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 34.7 atom percent B-10.	Prior to addition to SLC tank
SR 3.1.7.11	Verify the enrichment of boron in solution is ≥ 34.7 atom percent B-10.	24 months

NEW



△
D

Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

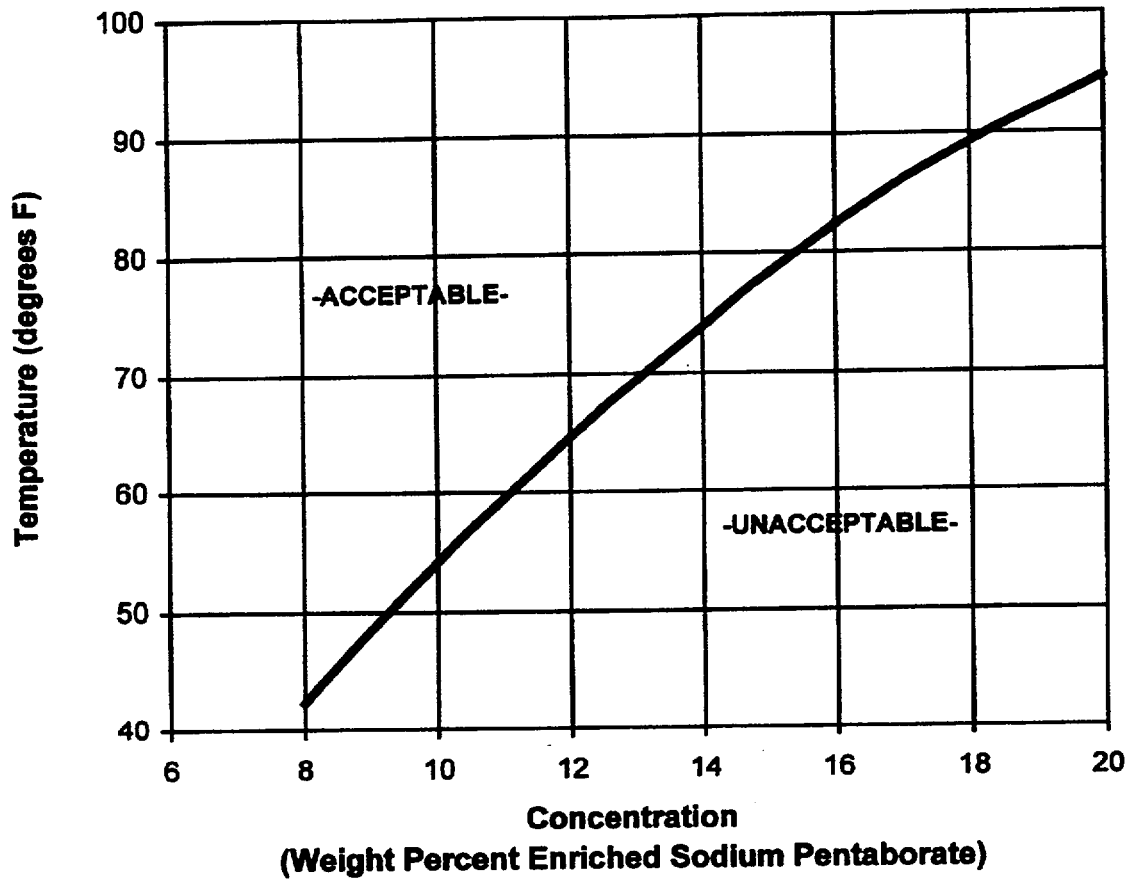


Figure 3.1.7-2 (page 1 of 1)
Sodium Pentaborate Solution
Temperature Versus Concentration Requirements

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram.

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core. A smaller tank containing demineralized water is provided for testing purposes.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 660 ppm of equivalent natural boron, in the reactor coolant at 70°F. To allow for potential leakage and imperfect mixing in the reactor system, an amount of boron equal to 125% of the amount cited above is injected (Ref. 2). The volume versus concentration limits in Figure 3.1.7-1 and the concentration versus temperature limits in Figure 3.1.7-2 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction level in the boron solution storage tank (6 inches above tank bottom). No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

ACTIONS

A.1

If one SLC subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the shutdown function. However, the overall reliability is reduced because a single failure in the

(continued)

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BASES

ACTIONS

A.1 (continued)

remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the control rods to shut down the reactor.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or severe transient occurring concurrent with the failure of the control rods to shut down the reactor.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank or in the pump suction

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3 (continued)

pipng. The temperature versus concentration curve of Figure 3.1.7-2 ensures that a 10°F margin will be maintained above the saturation temperature. The 24 hour Frequency is based on operating experience and has shown there are relatively slow variations in the measured parameters of volume and temperature.

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.6 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual valves in the SLC System flow path provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance 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 verification of valve alignment 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 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron in the storage tank is maintained per Figure 3.1.7-1. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. SR 3.1.7.5 must also be performed anytime the temperature is restored to within the limits of Figure 3.1.7-2, to ensure that no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7

Demonstrating that each SLC System pump develops a flow rate ≥ 50 gpm at a discharge pressure ≥ 1275 psig by recirculating demineralized water to the test tank ensures that pump performance has not degraded during the surveillance interval. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms pump and motor capability and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve primer assembly. The replacement primer assembly for the explosive valves shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.1.7.8 and SR 3.1.7.9 (continued)

valve pathway tested should be alternated such that both complete flow paths are tested every 48 months at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction inlet to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping is unblocked is to manually initiate the system, except the explosive valves, and pump from the storage tank to the test tank. Upon completion of this verification, the pump suction piping must be flushed with demineralized water to ensure piping between the storage tank and pump suction is unblocked.

The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. This is especially true in light of the temperature verification of this piping required by SR 3.1.7.3. However, if, in performing SR 3.1.7.3, it is determined that the temperature of this piping has fallen below the specified minimum, SR 3.1.7.9 must be performed once within 24 hours after the piping temperature is restored to within the limits of Figure 3.1.7-2.

SR 3.1.7.10

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.7.10 (continued)

tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used. A single isotopic test from a single batch can suffice as the required analysis for any number of mixings and additions from this batch. Certified vendor analytical test results may be used to satisfy this requirement.

SR 3.1.7.11

The B-10 enrichment of boron in solution in the SLC tank is only affected by the B-10 enrichment of tank additions. The requirements of SR 3.1.7.10 serve to assure that tank additions contain the proper enrichment. SR 3.1.7.11 requires periodic verification of the B-10 enrichment of the solution in the SLC tank, providing added assurance that the proper B-10 enrichment is maintained.

REFERENCES

1. 10 CFR 50.62.
 2. UFSAR, Section 3.9.4.
 3. 10 CFR 50.36(c)(2)(ii).
-
-

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

Scram Discharge Volume (SDV) Vent and Drain Valves

**MARKUP OF CURRENT TECHNICAL SPECIFICATIONS
(CTS)**

DISCUSSION OF CHANGES (DOCs) TO THE CTS

**NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
FOR LESS RESTRICTIVE CHANGES**

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

**JUSTIFICATION FOR DIFFERENCES (JFDs) FROM
NUREG-1433, REVISION 1**

MARKUP OF NUREG-1433, REVISION 1, BASES

**JUSTIFICATION FOR DIFFERENCES (JFDs) FROM
NUREG-1433, REVISION 1, BASES**

**RETYPE PROPOSED IMPROVED TECHNICAL
SPECIFICATIONS (ITS) AND BASES**

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

**Scram Discharge Volume (SDV) Vent and Drain
Valves**

**MARKUP OF CURRENT TECHNICAL
SPECIFICATIONS (CTS)**

Specification 3.18

JAFNPP

A1

3.3.A (draft)

4.3.A (draft)

2. Reactivity margin - inoperable control rods

2. Reactivity margin - inoperable control rods

a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to the Cold Shutdown condition within 24 hours and shall not be restarted unless (1) investigation has shown that the cause of the failure is not a failed control rod drive mechanism cockle housing, and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.

If investigation shows that the cause of control rod failure is a cracked cockle housing, or if this possibility cannot be ruled out, the reactor shall not be restarted until the affected control rod drive has been replaced or repaired.

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30 percent power. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day when operating above 30 percent power.

See 3.1.8, 3.1.11, 3.1.12, 3.1.13, 3.1.14, 3.1.15, 3.1.16, 3.1.17, 3.1.18, 3.1.19, 3.1.20, 3.1.21, 3.1.22, 3.1.23, 3.1.24, 3.1.25, 3.1.26, 3.1.27, 3.1.28, 3.1.29, 3.1.30, 3.1.31, 3.1.32, 3.1.33, 3.1.34, 3.1.35, 3.1.36, 3.1.37, 3.1.38, 3.1.39, 3.1.40, 3.1.41, 3.1.42, 3.1.43, 3.1.44, 3.1.45, 3.1.46, 3.1.47, 3.1.48, 3.1.49, 3.1.50, 3.1.51, 3.1.52, 3.1.53, 3.1.54, 3.1.55, 3.1.56, 3.1.57, 3.1.58, 3.1.59, 3.1.60, 3.1.61, 3.1.62, 3.1.63, 3.1.64, 3.1.65, 3.1.66, 3.1.67, 3.1.68, 3.1.69, 3.1.70, 3.1.71, 3.1.72, 3.1.73, 3.1.74, 3.1.75, 3.1.76, 3.1.77, 3.1.78, 3.1.79, 3.1.80, 3.1.81, 3.1.82, 3.1.83, 3.1.84, 3.1.85, 3.1.86, 3.1.87, 3.1.88, 3.1.89, 3.1.90, 3.1.91, 3.1.92, 3.1.93, 3.1.94, 3.1.95, 3.1.96, 3.1.97, 3.1.98, 3.1.99, 3.1.100

b. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days (these valves may be closed intermittently for testing under administrative control).

c. The status of the pressure and level alarms for each accumulator shall be checked once per week.

d. When it is initially determined that a control rod is incapable of normal insertion, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted, shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any operating condition during the remainder of the highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted. If Specification 3.3.A.1 and 4.3.A.1 are met, reactor startup may proceed.

Add: LCO 3.1.8

add Applicability

add ACTIONS A, B, C, 12

Amendment No. 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 59, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 70, 71, 72, 73, 74, 75, 76, 77, 78, 79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91, 92, 93, 94, 95, 96, 97, 98, 99, 100

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REVISION D

Specification 3.1.8

JAFNPP

~~3.3.A.2 (cont'd)~~

< See ITS 3.1.3 >

- b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.
- c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

~~3.3.A.2 (cont'd)~~

In accordance with the Inservice Test Program

A1

A2

- e. The scram discharge volume drain and vent valves shall be full-travel cycled at least ~~once per quarter to verify that the~~ ^{LA1} valves close in less than 30 seconds and to assure proper valve stroke and operation.

- f. An instrument check of control rod position indication shall be performed once/day.

< see ITS 3.1.3 >

- d. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

< See ITS 3.1.3; 3.1.5 >

- e. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met.
 - (1) When operating with two or more inoperable control rods in the Startup/Hot Standby or Run modes at $\leq 10\%$ rated thermal power, control rod patterns shall be equivalent to those prescribed by the Banked Position Withdrawal Sequence (BPWS) or else the inoperable control rods shall be separated by two or more operable control rods. If this condition is not met, restore compliance with the condition within 4 hours. Otherwise be in hot shutdown within the following 12 hours.

< see ITS 3.1.1; 3.1.3 >

[ACTION C]

- (2) If nine or more control rods are inoperable, be in hot shutdown within 12 hours.

AMD 255

(A1)

JAFNPP

3.3.C (cont'd)

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Control Rod Notch Position Observed	Average Scram Insertion Time (Seconds)
46	0.361
38	0.977
24	2.112
04	3.764

3. The maximum scram insertion time for 90 percent insertion of any operable control rod shall not exceed 7.00 sec.

See ITS: 3.1.4 and 3.1.3

4.3.C (cont'd)

2. At 16-week intervals, 10 percent of the operable control rod drives shall be scram timed above 950 psig. The same control rod drives should not be tested each interval. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

See ITS: 3.1.4

LI.

3. All control rods shall be determined operable by demonstrating the scram discharge volume drain and vent valves are:

Item	Frequency
[SR 3.1.8.1] Verified Open	Once per 31 Days
[SR 3.1.8.2] Cycled Fully Closed and Open	In accordance with the Inservice Testing Program
[SR 3.1.8.3] Verified to close within 30 seconds after receipt of an actual or simulated scram signal and open when the actual or simulated scram signal is reset.	Once per 24 Months

Specification 3.1.8

AI

JAFNPP

3.3 (cont'd)

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1 percent Δk . If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

E. If Specifications 3.3.C and D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold condition within 24 hr.

LI

4.3 (cont'd)

D. Reactivity Anomalies

During the Startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

See ITS: 3.1.2

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

Scram Discharge Volume (SDV) Vent and Drain Valves

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

ADMINISTRATIVE CHANGES

- A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).
- A2 CTS 4.3.A.2.e and 4.3.C.3.b both have surveillance requirements to fully cycle the scram discharge volume vent and drain valves. ITS SR 3.1.8.2 combines these tests into one surveillance. Since the two tests accomplish the same thing the combination of the two requirements is considered administrative. In addition, since the 92 day surveillance frequency in CTS 4.3.A.2.e is consistent with the Inservice Testing Requirements, the proposed Frequency is "In accordance with the Inservice Testing Program" which is consistent with the current wording in CTS 4.3.C.3.b.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

- LA1 The requirement in CTS 4.3.A.2.e concerning the scram discharge volume drain and vent valve closure time criteria (30 seconds) is proposed to be relocated to the Inservice Testing Program. The Requirement in ITS SR 3.1.8.2 to cycle each SDV vent and drain valve to the fully closed and fully open position in accordance with the Inservice Testing Program and proposed SRs 3.1.8.1 (verification the valves are open) and SR 3.1.8.3 (verification of closure time during an actual or simulated scram signal) are adequate to ensure the valves are OPERABLE. Testing of valves is required to be performed in accordance with Section XI of the ASME Code and applicable Addenda as required by 10 CFR 50.55a, except where relief has been requested. Therefore, the relocated requirement is not necessary to be in the ITS to provide adequate protection of the public health and safety. Changes to the testing Frequency and closure criteria of the Inservice Test Program will be controlled by the provisions of 10 CFR 50.59.

DISCUSSION OF CHANGES
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

- L1 CTS 4.3.A.2.b, CTS 4.3.A.2.e and CTS 4.3.C.3 include requirements for SDV vent and drain valves. These requirements are currently associated with control rod operability. The default action for CTS 4.3.A.2.b and 4.3.A.2.e is to be in Hot Shutdown (Mode 3) in 12 hours (CTS 3.3.A.2.e.2), while the default action for CTS 4.3.C.3 is to be in a cold condition within 24 hours (CTS 3.3.E). These default actions are not consistent. In the ITS, all the requirements for SDV vent and drain valves are included in one Specification for consistency. In ITS 3.1.8, the SDV vent and drain valves are only required to be Operable in MODES 1 and 2. In MODES 1 and 2, a scram may be required; therefore the SDV vent and drain valves must be Operable. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. Since a scram is not required in MODE 3 this change is acceptable. Therefore, the default action of CTS 3.3.A.2.e.2 is adopted as reflected in ITS 3.1.8 Action C. This change is consistent with NUREG-1433, Revision 1.
- L2 CTS 4.3.C.3 contains Surveillance Requirements for SDV vent and drain valves but the CTS do not provide specific actions if SDV vent and drain valves are inoperable. The primary safety function of the SDV vent and drain valves is to isolate the SDV during a scram to contain the reactor coolant leakage past the CRD seals. This isolation function can be satisfied with only one valve OPERABLE in each line or the line is isolated. Therefore, the actions are provided to:
- 1) Allow 7 days to isolate an inoperable SDV vent or drain valve provided at least one valve in each line is Operable (ITS 3.1.8 ACTION A).
 - 2) Establish an 8 hour limit when both valves in a line are inoperable and, allowing the option of isolating the line during this time (ITS 3.1.8 ACTION B).
 - 3) Require the plant to be placed in MODE 3 in 12 hours (ITS 3.1.8 Required Action C.1) if any Required Action and associated Completion Time is not met (See L1).
 - 4) Recognize that the SDV vent and drain valves are normally open to prevent accumulation of water in the SDV from leakage. Therefore, a Note is added to ITS 3.1.8 ACTIONS, allowing periodic opening of the affected line for draining and venting of the SDV. This will be necessary to avoid an automatic reactor scram on high level in the SDV.

DISCUSSION OF CHANGES
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

- 5) Provide a Note at the start of the ACTIONS Table (Separate Condition entry is allowed for each SDV vent and drain line) to provide more explicit instructions for proper application of the Actions for ITS 1.3, "Completion Times." Each SDV line is tested independently and allowed a specified period of time to confirm it isolated or capable of isolation, or restore the complete function of the line.

The time allowed to isolate an inoperable SDV vent or drain line, and the option to administratively unisolate an SDV line isolated by a Required Action are consistent with the BWR Standard Technical Specifications, NUREG-1433, Revision 1, as modified by the allowance to isolate rather than restore the line when one SDV vent and drain valve in one or more lines is inoperable. The SDV vent and drain valve's primary function is to isolate the SDV during a scram to contain the reactor coolant discharge. The isolation function is satisfied if the associated lines are isolated in the event one SDV vent or drain valve in one or more lines is inoperable. These increased allowances are deemed to not substantially increase the risk of a SDV failing to accept the control rod drive water displaced during a scram.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

**Scram Discharge Volume (SDV) Vent and Drain
Valves**

**NO SIGNIFICANT HAZARDS CONSIDERATION
(NSHC) FOR LESS RESTRICTIVE CHANGES**

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. 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 proposes to provide one Applicability for all Technical Specification requirements associated with scram discharge volume (SDV) vent and drain valves (MODES 1 and 2 instead of MODES 1, 2 and 3). In addition, a default condition has been provided consistent with the Applicability and current requirements in CTS 3.3.A.2.e.2 (Be in MODE 3 within 12 hours). The purpose of the SDV vent and drain valves is to isolate the SDV during a scram to contain the fluid released above the CRD piston to ensure that 10 CFR 100 limits are not exceeded. The SDV vent and drain valves are not identified as initiators for any accidents previously evaluated, therefore, this change will not increase the probability of accidents. During MODE 3, the control rods are already inserted, therefore there is no need to help ensure the scram function. During this mode of operation (with the control rods inserted), if a scram signal were generated, there is no displacement of water above the CRD piston, therefore 10 CFR 100 limits are of no concern. The consequences of an accident occurring in MODE 3 with the SDV drain and vent valves inoperable is bounded by the consequences of an accident in MODES 1 and 2. The default requirement to be in MODE 3 in 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Therefore, this change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not involve a physical modification to the plant or a new mode of operation and therefore does not create the possibility of a new or different kind of accident from any accident previously evaluated.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

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

This change proposes to provide one Applicability for all Technical Specification requirements associated with scram discharge volume (SDV) vent and drain valves (MODES 1 and 2 instead of MODES 1, 2 and 3). In addition, a default condition has been provided consistent with the Applicability and current requirements in CTS 3.3.A.2.e.2 (Be in MODE 3 within 12 hours). The purpose of the SDV vent and drain valves is to isolate the SDV during a scram to contain the fluid released above the CRD piston to ensure that 10 CFR 100 limits are not exceeded. During MODE 3, the control rods are already inserted, therefore, there is no need to help ensure the scram function. During this mode of operation (with the control rods inserted), if a scram signal were generated, there is no displacement of water above the CRD piston, therefore 10 CFR 100 limits are of no concern. Therefore, the consequences of an accident occurring in MODE 3 with the SDV drain and vent valves inoperable is bounded by the consequences of an accident in MODES 1 and 2. The default requirement to be in MODE 3 in 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The proposed Applicability will ensure the SDV vent and drain valves are Operable when required while the default action will ensure the plant leaves the Applicability in a orderly manner without challenging plant systems. Therefore, this change does not result in a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. 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 proposes to provide allowed outage times if one or two SDV vent and drain valves are inoperable. The purpose of the SDV vent and drain valves is to isolate the SDV during a scram to contain the fluid released above the CRD piston to ensure that 10 CFR 100 limits are not exceeded. If one valve is inoperable, the other valve accomplishes this function. This is done without substantially increasing the risk of a scram with an additional failure that could allow the SDV to remain unisolated. The probability of a scram occurring, significant CRD seal leakage, and a failure of the OPERABLE valve occurring is very remote in a 7 day period. If both valves are inoperable the line is required to be isolated within 8 hours. In addition, if allowed outage times are not met then the plant is required to be in MODE 3 in 12 hours. The allowed outage times provide enough time to accomplish this function in a planned manner without substantially increasing the risk of reactor coolant system leakage occurring due to a scram. The probability of a scram occurring while the line is not isolated with significant CRD seal leakage is low. The SDV vent and drain valves are not identified as initiators for any accidents previously evaluated. Therefore, this change will not increase the probability of accidents. The consequences of an accident occurring during the proposed allowed outage times are the same as the consequences of an accident occurring during the current time period associated with required shutdown actions in the same conditions. Therefore this change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not involve a physical modification to the plant or a new mode of operation and therefore does not create the possibility of a new or different kind of accident from any accident previously evaluated.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

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

This change will not significantly reduce a margin of safety. The SDV vent and drain valves contain the fluid released above the CRD piston to ensure 10 CFR 100 limits are not exceeded. Providing allowed outage time for inoperable SDV vent and drain valves is acceptable since the possibility of a scram occurring along with a single failure (for one valve inoperable) and significant CRD seal leakage is remote during the allowed outage times proposed. In addition, if allowed outage times are not met then the plant is required to be in MODE 3 in 12 hours. The safety analysis is unaffected because the current analysis assumptions are still being maintained. As such no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

**Scram Discharge Volume (SDV) Vent and Drain
Valves**

**MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION**

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

[L1] LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

[L1] APPLICABILITY: MODES 1 and 2.

ACTIONS

[L2] 1. Separate Condition entry is allowed for each SDV vent and drain line.

NOTE

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SDV vent or drain lines with one valve inoperable.	A.1 Restore valve to OPERABLE status. <i>Isolate the associated line.</i>	7 days
B. One or more SDV vent or drain lines with both valves inoperable.	B.1 NOTE 2. An isolated line may be unisolated under administrative control to allow draining and venting of the SDV. Isolate the associated line.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

X1

JAFNPP

BWR/4 STS

Amendment

Rev 1 / 04/07/95

Typ All Pages

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.8.1</p> <p>-----NOTE----- Not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. -----</p> <p>Verify each SDV vent and drain valve is open.</p> <p>[4.3.A.2.b] [4.3.C.3.a]</p>	<p>31 days</p>
<p>SR 3.1.8.2</p> <p>Cycle each SDV vent and drain valve to the fully closed and fully open position.</p> <p>[4.3.A.2.e] [4.3.C.3.b]</p>	<p>92 days</p> <p>In accordance with the Inservice Testing Program</p>
<p>SR 3.1.8.3</p> <p>Verify each SDV vent and drain valve:</p> <p>a. Closes in \leq [60] seconds after receipt of an actual or simulated scram signal; and</p> <p>b. Opens when the actual or simulated scram signal is reset.</p> <p>[4.3.C.3.c]</p>	<p>30 months</p> <p>24</p> <p>CLB1</p> <p>CLB2</p>

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

**Scram Discharge Volume (SDV) Vent and Drain
Valves**

**JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1**

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The Frequency of SR 3.1.8.2 has been changed from 92 days to "In accordance with the Inservice Testing Program" consistent with CTS 4.3.C.3.b.

CLB2 The brackets have been removed from the 18 month Frequency in SR 3.1.8.3 and it has been extended to 24 months consistent with CTS 4.3.C.3.c.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 The brackets have been removed and the proper plant specific value has been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 JAFNPP proposes to isolate the associated line when one valve is inoperable, instead of requiring the valve to be restored to Operable status. The SDV vent and drain valve's primary function is to isolate the SDV during a scram to contain the reactor coolant discharge. The isolation function is satisfied if the line is isolated. Therefore, Required Action A.1 has been changed to require the associated line to be isolated. In addition, the NOTE of Required Action B.1 has been moved so that it applies to both ACTION A and B. In both cases, it is necessary to unisolate the line under administrative controls to allow draining and venting of the SDV. This is done to prevent the SCRAM on "Scram Discharge Volume Water Level-High." This change has been approved by the NRC in the Safety Evaluation Report for Washington

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 (continued)

Nuclear Plant Unit 2 (WNP-2), Amendment 134 and LaSalle Units 1 and 2, Amendments 89 and 94, respectively. The JAFNPP design is similar to the WNP-2 and LaSalle design.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

**Scram Discharge Volume (SDV) Vent and Drain
Valves**

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. ~~The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series for a total of four vent valves.~~ The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

DBI
Each
has
each having

DBI
For a total of four drain valves
DBI
each having
Separate DBI

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scrambling. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

- a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and
- b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow continuous drainage of the SDV during normal plant operation

(continued)

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of the NRC Policy Statement. 10 CFR 50.36(c)(2)(i) (Ref 4) X1

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that only a single control rod can be withdrawn. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram. PA1

ACTIONS

The ACTIONS ^Ttable is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions PA2
PA3

(continued)

BASES

The ACTIONS Table is modified by a second Note stating that an isolated line may be unisolated under administrative control to allow draining and venting of the SDV

ACTIONS
(continued)

for each inoperable SDV line. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

the line must be isolated to contain the reactor coolant during a scram.

A.1

When one SDV vent or drain valve is inoperable in one or more lines, the valves must be restored to OPERABLE status within 7 days. The Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring while the valve(s) are inoperable. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion may not be preserved, and a higher risk exists to allow reactor water out of the primary system during a scram.

and the lines are not isolated

IS met PA2

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram.

When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. Required Action B.1 is modified by a Note that allows periodic draining and venting of the SDV when a line is isolated.

During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and unlikely of significant CRD seal leakage.

the PA2

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO

(continued)

BASES

ACTIONS

C.1 (continued)

does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position.

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

SR 3.1.8.2

*is in accordance with the
Inservice Testing Program requirements.*

CLB1

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The 92-day Frequency is based on operating experience and takes into account the level of redundancy in the system design.

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.3 (continued)

30 → 60 seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis (Ref. 2). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Handwritten annotations: (30), (DBI), (3), (CLB2), (24), (CLB2), (24), (CLB2), (and analysis), (PA4)

REFERENCES

1. (UFSAR, Section [4.2.3/2.7.3]) 3.5.5.2 (DB2)
2. 10 CFR 100.
3. NUREG-0803, Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping, August 1981. (PAS)

4. 10 CFR 50.36 (c)(2)(ii) (X1)

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

**Scram Discharge Volume (SDV) Vent and Drain
Valves**

**JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1, BASES**

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

RETENTION OF EXISTING REQUIREMENT (CLB)

- CLB1 The Frequency of SR 3.1.8.2 has been changed from 92 days to "In accordance with the Inservice Testing Program" consistent with CTS 4.3.C.3.b. The Bases has been changed to reflect this change.
- CLB2 The Frequency of SR 3.1.8.3 has been changed from 18 months to 24 months consistent with CTS 4.3.C.3.c. The Bases has been changed to reflect this change.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

- PA1 An error made between Revision 0 and Revision 1 to the NUREG BASES has been corrected.
- PA2 Editorial changes have been made to correct a grammatical/typographical error.
- PA3 Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.
- PA4 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.
- PA5 The quotations used in the Bases References have been removed. The Writer's Guide does not require the use of quotations.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

- DB1 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific design.
- DB2 The brackets have been removed from the references and the plant specific references have been provided.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.8 - SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

- X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CDR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.
- X2 JAFNPP proposes to isolate the associated line when one valve is inoperable, instead of requiring the valve to be restored to Operable status. The SDV vent and drain valve's primary function is to isolate the SDV during a scram to contain the reactor coolant discharge. The isolation function is satisfied if the line is isolated. Therefore, Required Action A.1 has been changed to require the associated line to be isolated. In addition, the NOTE of Required Action B.1 has been moved so that it applies to both ACTION A and B. In both cases, it is necessary to unisolate the line under administrative controls to allow draining and venting of the SDV. This is done to prevent the SCRAM on "Scram Discharge Volume Water Level-High." This change has been approved by the NRC in the Safety Evaluation Report for Washington Nuclear Plant Unit 2 (WNP-2), Amendment 134 and LaSalle Units 1 and 2, Amendments 89 and 94, respectively. The JAFNPP design is similar to the WNP-2 and LaSalle design.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.8

**Scram Discharge Volume (SDV) Vent and Drain
Valves**

**RETYPE PROPOSED IMPROVED TECHNICAL
SPECIFICATIONS (ITS) AND BASES**

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

LCO 3.1.8 Each SDV vent and drain valve shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTES-----

1. Separate Condition entry is allowed for each SDV vent and drain line.
 2. An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SDV vent or drain lines with one valve inoperable.	A.1 Isolate the associated line.	7 days
B. One or more SDV vent or drain lines with both valves inoperable.	B.1 Isolate the associated line.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.8.1</p> <p>-----NOTE----- Not required to be met on vent and drain valves closed during performance of SR 3.1.8.2. -----</p> <p>Verify each SDV vent and drain valve is open.</p>	<p>31 days</p>
<p>SR 3.1.8.2</p> <p>Cycle each SDV vent and drain valve to the fully closed and fully open position.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.1.8.3</p> <p>Verify each SDV vent and drain valve:</p> <p>a. Closes in \leq 30 seconds after receipt of an actual or simulated scram signal; and</p> <p>b. Opens when the actual or simulated scram signal is reset.</p>	<p>24 months</p>

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV is a volume of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two SDVs (headers) and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. Each instrument volume has a drain line each having two valves in series for a total of four drain valves. Each header is connected to a separate vent line each having two valves in series for a total of four vent valves. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE
SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all of the control rods are capable of scramming. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

- a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 100 (Ref. 2); and
- b. Open on scram reset to maintain the SDV vent and drain path open so that there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 100 (Ref. 2), and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves allow

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

continuous drainage of the SDV during normal plant operation to ensure that the SDV has sufficient capacity to contain the reactor coolant discharge during a full core scram. To automatically ensure this capacity, a reactor scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") is initiated if the SDV water level in the instrument volume exceeds a specified setpoint. The setpoint is chosen so that all control rods are inserted before the SDV has insufficient volume to accept a full scram.

SDV vent and drain valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The OPERABILITY of all SDV vent and drain valves ensures that the SDV vent and drain valves will close during a scram to contain reactor water discharged to the SDV piping. Since the vent and drain lines are provided with two valves in series, the single failure of one valve in the open position will not impair the isolation function of the system. Additionally, the valves are required to open on scram reset to ensure that a path is available for the SDV piping to drain freely at other times.

APPLICABILITY

In MODES 1 and 2, scram may be required; therefore, the SDV vent and drain valves must be OPERABLE. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. Also, during MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Therefore, the SDV vent and drain valves are not required to be OPERABLE in these MODES since the reactor is subcritical and only one rod may be withdrawn and subject to scram.

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each SDV vent and drain line. This is acceptable, since the Required Actions provide appropriate compensatory actions for each inoperable SDV line. Complying with the Required Actions may allow for

(continued)

BASES

ACTIONS
(continued)

continued operation, and subsequent inoperable SDV lines are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS Table is modified by a second Note stating that an isolated line may be unisolated under administrative control to allow draining and venting of the SDV. When a line is isolated, the potential for an inadvertent scram due to high SDV level is increased. During these periods, the line may be unisolated under administrative control. This allows any accumulated water in the line to be drained, to preclude a reactor scram on SDV high level. This is acceptable since the administrative controls ensure the valve can be closed quickly, by a dedicated operator, if a scram occurs with the valve open.

A.1

When one SDV vent or drain valve is inoperable in one or more lines the line must be isolated to contain the reactor coolant during a scram. The 7 day Completion Time is reasonable, given the level of redundancy in the lines and the low probability of a scram occurring while the valve(s) are inoperable and the lines are not isolated. The SDV is still isolable since the redundant valve in the affected line is OPERABLE. During these periods, the single failure criterion is not met, and a higher risk exists to allow reactor water out of the primary system during a scram.

B.1

If both valves in a line are inoperable, the line must be isolated to contain the reactor coolant during a scram. The 8 hour Completion Time to isolate the line is based on the low probability of a scram occurring while the line is not isolated and the unlikelihood of significant CRD seal leakage.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be

(continued)

BASES

ACTIONS

C.1 (continued)

brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.1

During normal operation, the SDV vent and drain valves should be in the open position (except when performing SR 3.1.8.2) to allow for drainage of the SDV piping. Verifying that each valve is in the open position ensures that the SDV vent and drain valves will perform their intended functions during normal operation. This SR does not require any testing or valve manipulation; rather, it involves verification that the valves are in the correct position.

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

SR 3.1.8.2

During a scram, the SDV vent and drain valves should close to contain the reactor water discharged to the SDV piping. Cycling each valve through its complete range of motion (closed and open) ensures that the valve will function properly during a scram. The Frequency is in accordance with the Inservice Testing Program requirements.

SR 3.1.8.3

SR 3.1.8.3 is an integrated test of the SDV vent and drain valves to verify total system performance. After receipt of a simulated or actual scram signal, the closure of the SDV vent and drain valves is verified. The closure time of 30 seconds after receipt of a scram signal is based on the bounding leakage case evaluated in the accident analysis

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.3 (continued)

(Ref. 3). Similarly, after receipt of a simulated or actual scram reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3 overlap this Surveillance to provide complete testing of the assumed safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience and analysis has shown these components pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 3.5.5.2.
 2. 10 CFR 100.
 3. NUREG-0803, Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping, August 1981.
 4. 10 CFR 50.36(c)(2)(ii).
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