Attachment 2:

- **3.4 – Vol. 11**
- **3.6 – Vol. 13 and 14**
- **3.7 – Vol. 15**

ATTACHMENT 2

REVISION E
ITS SECTION 3.4, SECTION 3.6, AND SECTION 3.7

SUMMARY OF CHANGES TO ITS SECTION 3.4

SUMMARY OF CHANGES TO ITS SECTION 3.4 - REVISION E

Source of Change	Summary of Change	Affected Pages
RAI 3.4-Generic (as modified)	The term "jet pump loop flow" was replaced by "recirculation loop jet pump flow" in SR 3.4.1.2 and associated Bases to be consistent with the NUREG-1433 terminology. The term "recirculation drive flow" was replaced by "recirculation pump flow" in SR 3.4.2.1 to be consistent with NUREG-1433 and the term "jet pump loop flow" was replaced by "recirculation loop jet pump flow" in SR 3.4.2.1 to be consistent with the terminology in ITS SR 3.4.1.1. In addition, while not identified in the RAI response, similar changes were done to ITS SR 3.4.9.4 and associated Bases, DOC, and NSHC.	**Specification 3.4.1** DOC M2 (DOCs p 1 of 3) ITS mark-up p 3.4-2 ITS Bases mark-up p B 3.4-5 Retyped ITS p 3.4-3 Retyped ITS Bases p B 3.4-7 **Specification 3.4.2** CTS mark-up p 2 of 3 DOC M2 (DOCs p 2 of 4) ITS mark-up p 3.4-4 ITS Bases mark-up p B 3.4-9 and B 3.4-10 Retyped ITS p 3.4-5 Retyped ITS Bases p B 3.4-11 and B 3.4-12 **Specification 3.4.9** DOC L2 (DOCs p 5 of 7, 6 of 7, and 7 of 7) NSHC L7 (NSHCs p 3 of 4 and 4 of 4) ITS mark-up p Insert Page 3.4-25 ITS Bases mark-up p Insert Page B 3.4-53 Retyped ITS p 3.4-21 Retyped ITS Bases p B 3.4-52

Source of Change	Summary of Change	Affected Pages
RAI 3.4.1-01	The ITS added a specific ACTION (ACTION B) for when the recirculation loop flows are not matched. ACTION B allowed 2 hours to restore matched flow or declare the loop with the lowest flow "not in operation." Since this specific ACTION was not in the CTS, the NRC believed this new requirement to be a beyond scope change. Therefore, this ACTION is being deleted and the general ACTION (new ACTION B, old ACTION C) will now provide the actions for when recirculation loop flows are not matched. The Bases will continue to state that when loop flows are not matched, then the loop with the lower flow must be declared "not in operation." In addition, with one loop not in operation, ACTION C provides 24 hours for the plant to establish the single loop requirements. Currently, 8 hours is provided.	**Specification 3.4.1** CTS mark-up p 1 of 1 DOCs M2 and L1 (DOCs p 2 of 3 and 3 of 3) NSHC L1 (NSHCs p 1 of 2 and 2 of 2) ITS mark-up p 3.4-1, Insert Page 3.4-1, and 3.4-2 JFDs CLB1 and X1 (deleted) (JFDs p 1 of 1) ITS Bases mark-up p B 3.4-3, B 3.4-4, Insert Page B 3.4-4, and B 3.4-5 Retyped ITS p 3.4-2 Retyped ITS Bases p B 3.4-4, B 3.4-5, B 3.4-6, and B 3.4-7
Amendment 267	Amendment 267 deleted CTS 3.6.E.5, which provided an allowance that during hydrostatic testing the safety and safety/relief valves did not have to be Operable. This allowance is now effectively in CTS 3.12.A (ITS 3.10.1).	**Specification 3.4.3** CTS mark-up p 1 of 3 (renumbered only, 2 of 3 (renumbered only), and 3 of 3 (also, old p 4 of 4, a blank page, was deleted) DOCs A5 (deleted) and LA4 (deleted) (DOCs p 1 of 4 and 3 of 4)
RAI 3.4.4-01	NUREG-1433 Required Action B.1 states "Reduce LEAKAGE to within limits." The ITS Required Action B.1 was modified to state "Reduce unidentified LEAKAGE increase to within limits." The change was justified by JFD PA1, which stated that the change was made to be consistent with the wording in Condition B. The NRC requested additional justification. JFD PA1 has been modified to provide additional justification, specifically, that ACTION B covers only an unidentified LEAKAGE increase while ACTION A covers all other types of LEAKAGE.	**Specification 3.4.4** JFD PA1 (JFDs p 1 of 1)
TSTF-205	TSTF-205 has been incorporated into the Bases for SR 3.4.5.2, the Channel Functional Test for the required Leakage Detection System instrumentation. The TSTF adds an clarification that, in lieu of testing all the required contacts of a channel relay, only a single contact need be tested (i.e., verify change of state of only a single contact).	**Specification 3.4.5** ITS Bases mark-up p B 3.4-32 and Insert Page B 3.4-32 Bases JFD TA2 (Bases JFDs p 2 of 2) Retyped ITS Bases p B 3.4-29

Source of Change	Summary of Change	Affected Pages
Amendment 261	This amendment modified the limits in CTS 4.6.C.1.d and e when sampling can be suspended and when a quantitative determination shall be made, respectively. However, CTS 4.6.C.1.d and e are being deleted (by DOC L5), thus the limit change has no impact on the ITS or Bases. Therefore, only the CTS mark-up pages have been updated to reflect the most current CTS page.	**Specification 3.4.6** CTS mark-up p 1 of 2 and 2 of 2
RAI 3.4.9-02	Condition A and Condition C have been modified to be consistent with the writer's guide and with other similar Conditions. Condition A now uses the term "MODE 1, 2, or 3," while Condition C now uses the term "other than MODES 1, 2, and 3."	**Specification 3.4.9** ITS mark-up p 3.4-23 and 3.4-24 ITS Bases mark-up p B 3.4-50 Retyped ITS 3.4-18 and 3.4-19 Retyped ITS Bases p B 3.4-49
RAI 3.4.9-04 (as modified)	The NRC noted that DOCs A2 and A4 seemed to be inconsistent with one another, with respect to the proper Applicability for the LCO and how it applied to the CTS. Therefore, DOCs A2 and A4 were revised to clear up this confusion.	**Specification 3.4.9** DOCs A2 and A4 (DOCs p 1 of 7)
RAI 3.4.9-05	CTS 4.6.A.1.a and b require recording the reactor vessel temperature when flange temperature is ≤ 120°F and ≤ 100°F, respectively, and the studs are tensioned. The Notes in ITS SRs 3.4.9.7 and 3.4.9.8 have the same temperature limits, but use MODE 4 in lieu of studs tensioned. The NRC requested better justification for changing from studs tensioned to MODE 4. After reviewing the change, JAFNPP has modified the SRs to maintain current licensing basis (i.e., studs tensioned), since the use of the term "MODE 4" could lead to ambiguity in application of the SR Notes.	**Specification 3.4.9** ITS mark-up p 3.4-26 ITS Bases mark-up p B 3.4-54 Retyped ITS p 3.4-22 Retyped ITS Bases p B 3.4-53
RAI 3.4.9-06 (as modified)	CTS 3.6.A.2, 3.6.A.3, and 3.6.A.4 specify operation being on or to the right of the curves in Figure 3.6-1 Part 1 or 2. The NRC noted that this requirement is not in the ITS and that no justification was provided for its deletion. The requirement has been moved to the Bases, both in the LCO section and in SR 3.4.9.1. A Discussion of Change has been provided for this relocation. The relocation is also consistent with previous ITS conversions (e.g., NMP2).	**Specification 3.4.9** CTS mark-up p 1 of 5 and 2 of 5 DOC LA5 (DOCs p 4 of 7 and 5 of 7) ITS Bases mark-up p B 3.4-52 Retyped ITS p B 3.4-51
RAI 3.4.9-07	CTS 4.6.A.6.a, b, and c specify that the differential temperatures be recorded and these requirements are not maintained in the ITS. The CTS mark-up indicates that these deletions are justified by DOC A6. However, the NRC noted that DOC A6 does not identify these three CTS as being changed and requested that DOC A6 be modified accordingly. DOC A6 has been modified to include these three CTS as being changed.	**Specification 3.4.9** DOC A6 (DOCs p 2 of 7)

Source of Change	Summary of Change	Affected Pages
Amendment 258	This amendment provided new PT Limit curves, one for up to 24 EFPY and one for 32 EFPY. The new curves have been adopted into the ITS.	**Specification 3.4.9** CTS mark-up p 1 of 5, 2 of 5, 4 of 5 and 5 of 5 DOC A5 (deleted) (DOCs p 1 of 7) ITS mark-up p 3.4-24, Insert Page 3.4-24, 3.4-26, Insert Page 3.4-26a, and Insert Page 3.4-26b ITS Bases mark-up p B 3.4-49, Insert Page B 3.4-49, B 3.4-50, Insert Page B 3.4-50, and Insert Page B 3.4-54 Retyped ITS p 3.4-20, 3.4-21, 3.4-23, and 3.4-24 Retyped ITS Bases p B 3.4-47, B 3.4-48, and B 3.4-54
Amendment 267	This amendment deleted the last paragraph of CTS 3.6.A.2, which provided an allowance that during hydrostatic testing the HPCI, RCIC, ADS, and S/RVs were not required to be Operable. This allowance is now effectively in CTS 3.12.A (ITS 3.10.1).	**Specification 3.4.9** CTS mark-up p 2 of 5 DOCs A9 (deleted) and LA4 (deleted) (DOCs p 2 of 7 and 4 of 7)

ITS CONVERSION PACKAGE

SECTION 3.4 - REACTOR COOLANT SYSTEM (RCS)

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

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

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

~~3.5 (cont'd)~~

[3.4.1] J. Recirculation Loops Operating ~~Thermal Hydraulic Stability~~

[APPLICABILITY] 1. When the reactor is in the ~~run mode:~~ MODES 1 and 2 (M1)

[LCO 3.4.1] a. Under normal operating conditions the reactor shall not be intentionally operated within the Power/Flow Exclusion Region defined in the Core Operating Limits Report (COLR).

[ACTION A] b. If the reactor has entered the Power/Flow Exclusion Region, the operator shall immediately insert control rods and/or increase recirculation flow to establish operation outside the region.

[3.4.1] K. Recirculation Loops Operating ~~Single Loop Operation~~

[Applicability] 1. The reactor may be started and operated, or reactor operation may continue, with a single Reactor Coolant System recirculation loop in operation. The requirements applicable to single-loop operation in Specifications ~~1.1.A~~ ~~2.1.A~~ 3.1.A, 3.1.B, ~~3.2.C~~ and 3.5.H shall be in effect within ~~8~~ hours or the reactor shall be placed in at least the hot shutdown mode within the following 12 hours.

[LCO 3.4.1]

[ACTION B]

[ACTION C]

~~2. During resumption of two-loop operation following a period of single-loop operation, the discharge valve of the lower speed pump shall not be opened unless the speed of the faster pump is less than 50 percent of its rated speed.~~

3. With no Reactor Coolant System recirculation loop in service, the reactor shall be placed in at least the hot shutdown mode within 12 hours.

[ACTION C]

A1

M2: Two recirculation loops with matched flows

A2: One or two loops

A2: One or two recirculation loops in operation

M2: add ACTIONS B & C for SR 3.4.1.2 not met

A3

24 L1

LA1

M3: Add SR 3.4.1.1

M2: Add SR 3.4.1.2

RA1 3.4.1-01

Page 1 of 1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (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 Although not stated CTS 3.5.J.1.a, Thermal Hydraulic Stability applies for both two loop and single loop operation. This clarification is reflected in ITS LCO 3.4.1 which requires operations to be outside the "Exclusion Region" of the power-to-flow map in both two loop and one loop operation. In addition, this clarification also applies to the current actions in CTS 3.5.J.1.b (proposed ITS 3.4.1 ACTION A). This change does not alter any technical requirements, it is therefore administrative and has no adverse impact on safety.

A3 The cross-reference to CTS 1.1.A in CTS 3.5.K.1 has been deleted since the proposed Safety Limit is applicable at all times. As currently written, the safety limit would not be required to be met for up to 8 hours after a single loop is in service. This is not the intent and would not be utilized in this manner. In addition, the reference to the APRM Flow Referenced Neutron Flux control rod block in CTS 2.1.A and in 3.2.C have been deleted since the function has been relocated from the Technical Specifications (see Discussion of Changes for LCO 3.3.2.1). Since the Safety Limit has the appropriate limit and since the APRM Flow Referenced Neutron Flux control rod block function has been relocated, no cross reference is needed therefore this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.5.J is applicable when the reactor is in the run mode. ITS 3.4.1 is applicable in MODES 1 and 2. This change is necessary since there is significant energy in the core in MODE 2 and postulated design basis accidents may occur in this condition. Since this change imposes operability requirements over a broader range of plant conditions, it is therefore more restrictive but necessary to ensure any postulated design basis accident will be bounded by the UFSAR analyses.

RAI 3.4-GEN

M2 A new requirement has been added to CTS 3.5.J.1.a (ITS LCO 3.4.1) which requires that the recirculation loop jet pump flow mismatch with both recirculation loops in operation be within the specified limits. The

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 (continued)

RAI 3.4.1-1

limits are included in proposed SR 3.4.1.2 and the surveillance must be performed within 24 hours after both loops are in operation and 24 hours, thereafter. In addition, ITS 3.4.1 ACTIONS B and C have been added to ensure that appropriate actions are taken when the flow mismatch limits are not met. This change imposes additional restrictions and is therefore is considered more restrictive on plant operations, but is necessary since the LOCA analysis conservatively assumes the break occurs in the loop with higher flow.

M3 ITS SR 3.4.1.1 has been added to CTS 3.5.J and 3.5.K to verify operation is outside the "Exclusion Region" of the power-to-flow map specified in the COLR every 12 hours. ITS SR 3.4.1.1 ensures the reactor THERMAL POWER and core flows are within appropriate parameter limits to prevent uncontrolled power oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal hydraulic instability. In addition, a Note is included which states that this SR is only required to be performed in MODE 1 because during plant operation in MODE 2 the APRM Neutron Flux-High (Startup) Function of ITS 3.3.1.1 will prevent entry into the "Exclusion Region." This is an added requirement and therefore, is considered more restrictive.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The requirements in CTS 3.5.K.2 that during resumption of two-loop operation following a period of single-loop operation, the discharge valve of the lower speed pump not be opened unless the speed of the faster pump is less than 50 percent of its rated speed is proposed to be relocated to the Technical Requirements Manual (TRM). The pump speed limit is considered an operational limit because it is not directly related to the ability of the system to perform its safety analysis functions. The pump speed is restricted to maintain reactor vessel internals vibration to within acceptable limits. These requirements are oriented toward maintaining long term Operability of the recirculation loops and do not necessarily have an immediate impact on their current Operability. Therefore, this relocated requirement is not required to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference in to the JAFNPP UFSAR at ITS implementation. Changes to the relocated requirements in the TRM will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

RAI 3.4.1-1

L1 ITS 3.4.1 ACTION B allows the requirements of the LCO to not be met for reasons other than Condition A (i.e., thermal hydraulic stability) for up to 24 hours. In this same condition, CTS 3.11.A, "APLHGR," and CTS 3.5.K.1, "Single Loop Operation," would require restoration of requirements within 8 hours or would require a plant shutdown within the following 12 hours. This change relaxes the effective allowed outage time to 24 hours to comply with the LCO when the reason for non-compliance is not related to thermal hydraulic stability. This LCO failure is essentially failing to comply with the appropriate modificaitons for single-loop operation. Relaxing the time to complete the limit modifications for single loop operation or to restore two loop operation in this condition is reasonable considering the low probability of an accident occurring during this period, the time required to perform the Required Action, and the frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected. The consequences of an accident are unchanged by adding additional time to complete limit modifications for single loop operation or to restore compliance with the LCO. Also, allowing this extended time will potentially avoid a plant transient caused by a plant shutdown. As such, this change does not represent a significant decrease in safety.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

RAI 3.4.1-1

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change relaxes the allowed outage time to 24 hours to comply with the LCO when the reason for non-compliance is not related to thermal hydraulic stability. The proposed change does not increase the probability of an accident. The time allowed to restore a second recirculation loop to operation or to satisfy single recirculation loop operation limits is not assumed in the initiation of an analyzed event. The change does not allow continuous operation but provides a time period which is acceptably short taking into consideration the small probability of an event occurring when a second recirculation loop is not operating and single loop operation limits are not met. Allowing additional time to comply with the LCO will not significantly increase the consequences of an accident. The consequences of an event occurring will be the same for proposed time periods as for the current time. This change will not alter assumptions relative to the mitigation of an accident or transient event. 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 change relaxes the allowed outage time to 24 hours to comply with the LCO. This change will not physically alter the plant (no new or different type 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

RAI 3.4.1-1

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

This change relaxes the allowed outage time to 24 hours to comply with the LCO when the reason for non-compliance is not related to thermal hydraulic stability. The increased time allowed to restore the second recirculation loop or to satisfy single recirculation loop operation limits is acceptable based on the small probability of an event occurring requiring recirculation loop operation to be within limits and the desire to minimize plant transients. While recirculation loop operation with matched flows is assumed in the LOCA analysis, allowing additional time to comply with the LCO does not significantly decrease the margin of safety. Also, the change provides the benefit of potentially avoiding a plant shutdown transient. The change allows more time to comply with the LCO instead of having to shut down. A plant shutdown is considered a transient due to the thermal effects it has on plant equipment. Therefore, this change does not involve a significant reduction in a margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

CLB1

[3.5.J.1.a] LCO 3.4.1 Two recirculation loops with matched flows shall be in operation and the reactor operating at core flow and THERMAL POWER conditions outside the Exclusion Region of the power-to-flow map specified in the COLR.

OR

CLB1 shall

One recirculation loop may be in operation provided with the following limits are applied when the associated LCO is applicable:

[3.5.K.1]

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits {specified in the COLR}; CLB3

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits {specified in the COLR}; and CLB3

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors, ~~Flow Biased Simulated Thermal Power—High~~ Neutron Flux-High (Flow Biased)), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation; and

d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor - Upscale), Allowable Value of Table 3.3.2.1-1 is reset for single loop operation. CLB3

[3.5.J.1] [3.5.K.1] APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
CLB1 A. Requirements of the LCO not met. B [3.5.K.1]	0.1 B Satisfy the requirements of the LCO.	24 hours

~~*Pending resolution of stability issue.~~ CLB1

for reasons other than Condition A

(continued)

[3.5.J.1] Insert ACTION A CLB1

RAI 3.4.1-1

Insert ACTION A

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two recirculation loop(s) in operation with core flow and THERMAL POWER conditions within the Exclusion Region of the power-to-flow map.	A.1 Initiate action to exit the Exclusion Region.	Immediately

RAI 3.4.1-1

CLB1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
CLB1 C [3.5.K.1] B. Required Action and associated Completion Time of Condition A not met.	C CLB1 B.1 Be in MODE 3. or B CLB1	12 hours
OR		
[3.5.K.3] No recirculation loops in operation.		

RAI 3.4.1-1

[M3] Insert SR 3.4.1.1 X2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.1.1 X2 2 ----------NOTE---------- Not required to be performed until 24 hours after both recirculation loops are in operation. ----------	
[M2] Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is:	24 hours
a. ≤ {10}% of rated core flow when operating at < {70}% of rated core flow; and	DB1
b. ≤ {5}% of rated core flow when operating at ≥ {70}% of rated core flow.	DB1

RAI 3.4-GEN

X2

Insert SR 3.4.1.1

SR 3.4.1.2	----------NOTE---------- Only required to be performed in MODE 1. ----------	
	Verify reactor operating at core flow and THERMAL POWER conditions outside the Exclusion Region of the power-to-flow map specified in the COLR.	12 hours

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

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

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The "Recirculation Loops Operating" Specification has been revised to reflect Current Licensing Basis requirements related to core thermal hydraulic stability. The Actions and Surveillances have been renumbered, where applicable to reflect this change.

RAI 3.4.1-1

CLB2 Not used.

CLB3 The brackets have been removed and the proper plant specific information/value has been provided consistent with the current requirements.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Editorial change have been made for enhanced clarity.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

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

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

RAI 3.4.1-1

X1 Not used.

X2 SR 3.4.1.1 has been added to ensure that operation is outside the "Exclusion Region" of the power-to-flow map. This ensures the requirements of the LCO are verified at the specified Frequency. Subsequent SRs have been renumbered, as applicable.

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Coolant Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Coolant Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one variable speed motor driven recirculation pump, a motor generator (MG) set to control pump speed, and associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core. The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat

(continued)

BASES

BACKGROUND (continued)

is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 45 to 100% of RTP) without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The MG set provides regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Coolant Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement. The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the FSAR.

(continued)

CLB1

Insert BKGD

The recirculation flow also provides sufficient core flow to ensure thermal-hydraulic stability of the core is maintained.

BASES

APPLICABLE SAFETY ANALYSES (continued)

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR setpoints for single loop operation are specified in the COLR. The APRM (Flow-Biased) simulated THERMAL POWER setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement.

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.2 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.2 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Flow Biased Simulated Thermal Power—High setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the assumptions of Reference 3.

(continued)

CLB1

Insert ASA

Operation of the Reactor Water Recirculation System also ensures adequate core flow at higher power levels such that conditions conducive to the onset of thermal hydraulic instability are avoided. The Updated Final Safety Analysis Report (UFSAR) Section 16.6 (Ref. 4) requires protection of fuel thermal safety limits from conditions caused by thermal hydraulic instability. Thermal hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit. The MCPR Safety Limit is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). Implementation of operability requirements for avoidance of, and protection from thermal-hydraulic instability, consistent with the BWR Owners' Group Long-Term Stability Solution Option I-D (Refs. 6 and 7) provides assurance that power oscillations are either prevented or can be readily detected and suppressed without exceeding the specified acceptable fuel design limits. To minimize the likelihood of thermal- hydraulic instability which results in power oscillations, a power-to-flow "Exclusion Region" is calculated using the approved methodology specified in Specification 5.6.5 The resulting "Exclusion Region" may change each fuel cycle and is therefore specified in the COLR. Entries into the "Exclusion Region" may occur as a result of an abnormal event, such as a single recirculation pump trip, loss of feedwater heating, or be required to prevent equipment damage.

The core-wide mode of oscillation in the neutron flux is more readily detected (and suppressed) than the regional mode of oscillation due to the spatial averaging of the Average Power Range Monitor (APRM). The Option I-D analysis for JAFNPP (Ref. 8) demonstrates that this protection is provided at a high statistical confidence level for regional mode oscillations. Reference 8 also demonstrates that the core-wide mode of oscillation is more likely to occur rather than regional oscillations due to the large single-phase pressure drop associated with the small fuel inlet orifice diameters.

PA3

Insert LCO

In addition, during two-loop and single-loop operation, the combination of core flow and THERMAL POWER must be outside the Exclusion Region of the power-to-flow map specified in the COLR to ensure core thermal-hydraulic instability does not occur.

BASES (continued)

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 24 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

(continued)

Insert A.1

A.1

With the reactor operating at core flow and THERMAL POWER conditions within the Exclusion Region of the power-to-flow map it is in a condition where thermal-hydraulic instabilities are conservatively predicted to occur, and must be brought to an operating state where such instabilities are not predicted to occur. To achieve this status, action must be taken immediately to exit the Exclusion Region. This is accomplished by inserting control rods or increasing core flow such that the combination of THERMAL POWER and core flow move to a point outside the Exclusion Region. The action is considered sufficient to preclude core thermal-hydraulic instabilities which could challenge the MCPR safety limit. The starting of a recirculation pump is not used as a means to exit the Exclusion Region of the power-to-flow map. Starting an idle recirculation pump could result in a reduction in inlet core enthalpy and enhance conditions necessary for thermal-hydraulic instabilities.

EDIT # (RAI 3.4.1-1)

BASES

ACTIONS (continued)

A.1

With no recirculation loops in operation or the Required Action and associated Completion Time of Condition A 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. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. 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 3.4.1.1

Insert SR 3.4.1.1

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

must be delayed "not in operation". (However, for the purpose of performing SR 3.4.1.1 the flow rate of both loops shall be used.)

(continued)

Insert SR 3.4.1.1

SR 3.4.1.1

This SR ensures the combination of core flow and THERMAL POWER are within appropriate limits to prevent uncontrolled thermal-hydraulic oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal-hydraulic instability. The power-to-flow map specified in the COLR is based on guidance provided in Reference 8. The 12 hour Frequency is based on operating experience and the operator's knowledge of the reactor status, including significant changes in THERMAL POWER and core flow.

This SR is modified by a Note that requires this surveillance to be performed only in MODE 1 because the APRM Neutron-Flux (Startup) High Function in LCO 3.3.1.1 will prevent operation in the Exclusion Region.

BASES (continued)

PA1

REFERENCES

1. FSAR, Section ~~[5.3.3.1]~~. 14.6 DB2
2. FSAR, Section ~~[5.5.1.1]~~. 14.5
3. ~~[Plant specific analysis for single loop operation.]~~

Insert Ref DB2

DB2

Insert Ref

REFERENCES 3. NEDO-24281, FitzPatrick Nuclear Power Plant Single-Loop Operation, August 1980.

4. UFSAR, Section 16.6

5. 10 CFR 50.36(c)(2)(ii).

6. NEDO-31960-A, BWR Owners' Group Long Term Stability Solutions Licensing Methodology, June 1991.

7. NEDO-31960-A, Supplement 1, BWR Owners' Group Long-Term Stability Solutions Licensing Methodology, March 1992.

8. GENE-637-004-0295, Application Of The "Regional Exclusion With Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) To The James A. FitzPatrick Nuclear Power Plant, February 1995.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

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

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.1 - RECIRCULATION LOOPS OPERATING

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The Bases have been revised to reflect the final resolution of the stability issue for JAFNPP and the existing requirements in CTS 3.5.J and 3.5.K. Subsequent Required Actions have been renumbered, as required.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Bases have been modified to reflect plant specific nomenclature.

PA2 Editorial changes have been made for clarification, correction, or improvement with no change in intent.

PA3 Bases have been modified to reflect changes made to the Specifications.

PA4 Bases have been modified to be consistent with other places in the Bases.

PA5 Bases have been modified to match the LCO.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 Changes have been made to reflect the plant specific values.

DB2 The Bases have been revised to reflect the appropriate JAFNPP References.

DB3 The brackets have been removed and the proper pant specific number included.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

RAI 3.4.1-1

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.

X2 Not used.

X3 The Bases description of SR 3.4.1.1 is added to reflect this new requirement added in accordance with M3. The subsequent Surveillance has been renumbered.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.1

Recirculation Loops Operating

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation and the reactor operating at core flow and THERMAL POWER conditions outside the Exclusion Region of the power-to-flow map specified in the COLR.

OR

One recirculation loop shall be in operation and the reactor operating at core flow and THERMAL POWER conditions outside the Exclusion Region of the power-to-flow map specified in the COLR with the following limits applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR; and

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Neutron Flux-High (Flow Biased)), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation; and

d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor-Upscale), Allowable Value of Table 3.3.2.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two recirculation loop(s) in operation with core flow and THERMAL POWER conditions within the Exclusion Region of the power-to-flow map.	A.1 Initiate action to exit the Exclusion Region.	Immediately
B. Requirements of the LCO not met for reasons other than Condition A.	B.1 Satisfy the requirements of the LCO.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR No recirculation loops in operation.	C.1 Be in MODE 3.	12 hours

RA1 3.4.1-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor operating at core flow and THERMAL POWER conditions outside the Exclusion Region of the power-to-flow map specified in the COLR.	12 hours
SR 3.4.1.2	----------NOTE---------- Not required to be performed until 24 hours after both recirculation loops are in operation. ---------- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours

RAI 3.4-GEN

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Water Recirculation System is designed to provide forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Water Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one variable speed motor driven recirculation pump, driven by a motor generator (MG) set to control pump speed, and associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core. The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat

(continued)

BASES

BACKGROUND (continued)

is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the void negative reactivity effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 45 to 100% of RTP) without having to move control rods and disturb desirable flux patterns. The recirculation flow also provides sufficient core flow to ensure thermal-hydraulic stability of the core is maintained.

Each recirculation loop is manually started from the control room. The MG set provides regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Water Recirculation System is an initial condition assumed in the Design Basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement. The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 14 of the UFSAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 14 of the UFSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) and the control rod block instrumentation Allowable Values are also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM Neutron Flux-High (Flow Biased) Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." The Rod Block Monitor-Upscale Allowable Value is specified in LCO 3.3.2.1, "Control Rod Block Instrumentation."

Operation of the Reactor Water Recirculation System also ensures adequate core flow at higher power levels such that conditions conducive to the onset of thermal hydraulic instability are avoided. The Updated Final Safety Analysis Report (UFSAR) Section 16.6 (Ref. 4) requires protection of fuel thermal safety limits from conditions caused by thermal hydraulic instability. Thermal hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit. The MCPR Safety Limit is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). Implementation of operability requirements for avoidance of, and protection from thermal-hydraulic instability, consistent with the BWR Owners' Group Long-Term Stability Solution Option I-D (Refs. 6 and 7) provides assurance that power oscillations are either prevented or can be readily detected and suppressed without exceeding the specified

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

acceptable fuel design limits. To minimize the likelihood of thermal- hydraulic instability which results in power oscillations, a power-to-flow "Exclusion Region" is calculated using the approved methodology specified in Specification 5.6.5 The resulting "Exclusion Region" may change each fuel cycle and is therefore specified in the COLR. Entries into the "Exclusion Region" may occur as a result of an abnormal event, such as a single recirculation pump trip, loss of feedwater heating, or be required to prevent equipment damage.

The core-wide mode of oscillation in the neutron flux is more readily detected (and suppressed) than the regional mode of oscillation due to the spatial averaging of the Average Power Range Monitor (APRM). The Option I-D analysis for JAFNPP (Ref. 8) demonstrates that this protection is provided at a high statistical confidence level for regional mode oscillations. Reference 8 also demonstrates that the core-wide mode of oscillation is more likely to occur rather than regional oscillations due to the large single-phase pressure drop associated with the small fuel inlet orifice diameters.

Recirculation loops operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.2 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.2 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), APRM Neutron Flux-High (Flow-Biased)—High Allowable Value (LCO 3.3.1.1) and the Rod Block Monitor—Upscale Allowable Value (LCO 3.3.2.1) must be applied to allow continued operation consistent with the assumptions of Reference 3. In addition, during two-loop and single-loop operation, the combination of core flow and THERMAL POWER must be outside the Exclusion Region of

RAI 3.4.1-01

(continued)

BASES

LCO (continued)

the power-to-flow map specified in the COLR to ensure core thermal-hydraulic instability does not occur.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Water Recirculation Water System are necessary since there is considerable energy in the reactor core, core thermal-hydraulic instability may occur, and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1

With the reactor operating at core flow and THERMAL POWER conditions within the Exclusion Region of the power-to-flow map it is in a condition where thermal-hydraulic instabilities are conservatively predicted to occur, and must be brought to an operating state where such instabilities are not predicted to occur. To achieve this status, action must be taken immediately to exit the Exclusion Region. This is accomplished by inserting control rods or increasing core flow such that the combination of THERMAL POWER and core flow move to a point outside the Exclusion Region. The action is considered sufficient to preclude core thermal-hydraulic instabilities which could challenge the MCPR safety limit. The starting of a recirculation pump is not used as a means to exit the Exclusion Region of the power-to-flow map. Starting an idle recirculation pump could result in a reduction in inlet core enthalpy and enhance conditions necessary for thermal-hydraulic instabilities.

B.1

With the requirements of the LCO not met for reasons other than Condition A the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the

←RAI 3.4.1-01

(continued)

BASES

ACTIONS

B.1 (continued)

pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

RAI 3.4.1-01

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS and control rod block Allowable Values, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

RAI 3.4.1-01

The 24 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

RAI 3.4.1-01

C.1

With any Required Action and associated Completion Time of Condition A or B not met, or no recirculation loop is in operation, 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. In this condition,

(continued)

BASES

ACTIONS

C.1 (continued)

the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. 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.

RAI 3.4.1-01

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the combination of core flow and THERMAL POWER are within appropriate limits to prevent uncontrolled thermal-hydraulic oscillations. At low recirculation flows and high reactor power, the reactor exhibits increased susceptibility to thermal-hydraulic instability. The power-to-flow map specified in the COLR is based on guidance provided in Reference 8. The 12 hour Frequency is based on operating experience and the operator's knowledge of the reactor status, including significant changes in THERMAL POWER and core flow.

This SR is modified by a Note that requires this surveillance to be performed only in MODE 1 because the APRM Neutron-Flux (Startup) High Function in LCO 3.3.1.1 will prevent operation in the Exclusion Region.

SR 3.4.1.2

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

RAI 3.4-GEN

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

Condition B must be entered, and the loop with the lower flow must be declared "not in operation". (However, for the purpose of performing SR 3.4.1.1, the flow rate of both loops shall be used.) The SR is not required when only one loop is in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1. UFSAR, Section 14.6.

2. UFSAR, Section 14.5.

3. NEDO-24281, FitzPatrick Nuclear Power Plant Single-Loop Operation, August 1980.

4. UFSAR, Section 16.6

5. 10 CFR 50.36(c)(2)(ii).

6. NEDO-31960-A, BWR Owners' Group Long Term Stability Solutions Licensing Methodology, June 1991.

7. NEDO-31960-A, Supplement 1, BWR Owners' Group Long-Term Stability Solutions Licensing Methodology, March 1992.

8. GENE-637-044-0295, Application Of The "Regional Exclusion With Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) To The James A. FitzPatrick Nuclear Power Plant, February 1995.

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.2

Jet Pumps

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

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.2

Jet Pumps

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.4.2 (A1)

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

See CTS: 3/4.6.F

4.6 (cont'd)

F. Structural Integrity

1. Nondestructive inspections shall be performed on the ASME Boiler and Pressure Vessel Code Class 1, 2 and 3 components and supports in accordance with the requirements of the weld and support inservice inspection program. This inservice inspection program is based on an NRC approved edition of, and addenda to, Section XI of the ASME Boiler and Pressure Vessel Code which is in effect 12 months or less prior to the beginning of the inspection interval.

2. An augmented inservice inspection program is required for those high stressed circumferential piping joints in the main steam and feedwater lines larger than 4 inches in diameter, where no restraint against pipe whip is provided. The augmented in-service inspection program shall consist of 100 percent inspection of these welds per inspection interval.

3. An Inservice Inspection Program for piping identified in the NRC Generic Letter 88-01 shall be implemented in accordance with NRC staff positions on schedules, methods, personnel, and sample expansion included in this Generic Letter, or in accordance with alternate measures approved by the NRC staff.

[Applicability]

[3.4.2] Jet Pumps

[LCO 3.4.2] Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the reactor shall be placed in a cold condition within 24 hours.

[ACTION A]

(L1) add SR 3.4.2.1 Notes

(12) (M1) MODE 3

[3.4.2] Jet Pumps

[SR 3.4.2.1]

Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

SR 3.4.2.1 (A2)

Amendment No. ~~96~~, ~~124~~, ~~140~~, ~~160~~, 203

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Specification 3.4.2

A1

RM 3.4-GEN

M2

[SR 3.4.2.1.a]

to speed ratio differs by ≤5% from the established patterns, and recirc loop jet pump flow to recirculation pump speed ratio differs by ≤ 5% from the established patterns

4.6 (cont'd)

1. The ~~two~~ recirculation ~~loops have a~~ pump flow ~~imbalance of 10 percent or more when the pumps are operated at the same speed~~.

2. ~~The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10 percent.~~

3. [SR 3.4.2.1.b] The diffuser to lower plenum differential pressure reading on an individual jet pump varies ~~from the average of all jet pump differential pressures~~ from established patterns by more than ~~10~~ 20 percent.

A3

L2

A. Whenever the reactor is in the startup/hot standby or run modes, and there is one loop recirculation flow, jet pump operability shall be verified as follows.

LA1

a. Baseline readings will be taken and operating characteristics for the following parameters established:

1. Jet Pump Loop Flow and Recirculation Pump Speed for the operating loop.

2. Individual Jet Pump percent differential pressures for all jet pumps.

L1

add SR 3.4.2.1 Notes

[SR 3.4.2.1]

b. Initially, and daily thereafter, jet pump operability will be verified by assuring that the following do not occur simultaneously:

A2

SR 3.4.2.1

Page 2 of 3

Specification 3.4.2 A1

M2 and recirculation pump flow

4.6 (cont'd)

[SR 3.4.2.1.a]

1. The ratio of jet pump loop flow to recirculation pump speed for the operating loop does not vary from the initially established value by more than 10 percent.

M.2 5

[SR 3.4.2.1.b]

2. The ratio of individual jet pump percent differential pressure to the loop's average jet pump percent differential pressure does not vary from the initially established value by more than 20 percent.

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DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.4.2 - JET PUMPS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (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 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 The wording in CTS 4.6.G and 4.6.G.A.b (ITS SR 3.4.2.1) was changed to require verification that one of the criteria be met, rather than require verification that none of the conditions exist simultaneously. This change is consistent with NUREG-1433, Revision 1, which is written in a positive mode, such that conditions must exist, rather than not exist. Since this change does not modify any technical requirements, it is therefore administrative.

A3 CTS 4.6.G.3 places requirements on individual jet pump differential pressure variation from the average of all jet pump differential pressures. ITS SR 3.4.2.1 places requirements on individual jet pump differential pressure variation from established patterns. This change is consistent with the recommendations provided in General Electric Service Information Letter (SIL) No. 330, Jet Pump Beam Cracks," and NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure." Since the jet pump diffuser to lower plenum differential pressure or relationship of one jet pump to the loop average is repeatable, both methods of comparison are considered equivalent. Therefore, this change is considered administrative. In addition, the wording is consistent with CTS 4.6.G.A.b.2.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.6.G requires that, if a jet pump is determined to be inoperable, that the reactor be placed in Cold Shutdown within 24 hours. ITS 3.4.2 requires that, if one or more jet pumps are inoperable, the plant be placed in MODE 3 in 12 hours. CTS 3.0.A states that "Limiting Conditions for Operation and Action requirements shall be applicable during the Operational Conditions (modes) specified for each specification." CTS 3.6.G is applicable in the Startup/Hot Standby and Run MODES; therefore, the requirement to place the plant in Cold Shutdown is not applicable after reaching the Hot Shutdown mode. The ITS action requires the plant to be placed in MODE 3, which is outside the MODE of applicability, within 12 hours. This change imposes an

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 (continued)

additional restriction on plant operation, which therefore constitutes a more restrictive change, and has no adverse impact on safety.

M2 CTS 4.6.G requires that certain conditions do not occur simultaneously. Two of these conditions are, (1) the two recirculation loops have a flow imbalance of ≥ 10% when the pumps are operated at the same speed, and (2) the indicated value of core flow rate varies from the value derived from loop flow measurements by > 10%. ITS SR 3.4.2.1 specifies one condition that may be used to verify operability of the jet pumps to be, "recirculation pump flow to speed ratio and recirculation loop jet pump flow to recirculation pump speed ratio both differ by ≤ 5% from established patterns." This change imposes new requirements on recirculation pump flow to speed ratio and recirculation loop jet pump flow to recirculation pump speed ratio. These requirements are designed to allow detection of a change in the relationship between recirculation pump speed, recirculation loop flow, and recirculation loop jet pump flow. A change in these relationships could indicate a plug, flow restriction, loss in pump hydraulic performance, leakage, or a new flow path between the recirculation pump discharge and a jet pump nozzle. These two Surveillances are designed as a more meaningful method of detection of significant degradation in jet pump performance prior to jet pump failure. This change is consistent with the type of surveillances in CTS 4.6.G.A.b.1 and 4.6.G.A.b.2 for single loop operation and therefore the surveillances are combined for both single and two loop operation (proposed SR 3.4.2.1). The existing criteria in CTS 4.6.G.A.b.1 if the jet pump flows differs from the established pattern by more than 10% has also been reduced to 5%. These changes impose new requirements on jet pump operability, which constitutes a more restrictive change, and has no adverse impact on safety.

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TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details in CTS 4.6.G.A to obtain base line data for single loop operation is proposed to be relocated to the Bases. The requirement to perform proposed SR 3.4.1.1 is adequate to ensure jet pump operability is evaluated at the specified frequency. Without baseline data the evaluation cannot be properly performed since the pattern may change after each fuel assembly replacement or shuffle, as well as in any modifications to fuel support orifice size or core bypass flow. Therefore it will be prudent that new patterns be established when

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 (continued)

changes have been made and during single loop operation. Therefore the relocated requirement is 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 proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 4.6.G and 4.6.G.A.b are revised by adopting two Notes which relax the Frequency by allowing a 4 hour delay to perform the Surveillance after the associated recirculation loop is in operation (ITS SR 3.4.2.1 Note 1), and a delay in performance of the Surveillance until 24 hours after the plant exceeds 25% RTP (ITS SR 3.4.2.1 Note 2). This is a relaxation of requirements, which is less restrictive. This change is acceptable for the following reasons. The first Note permits a delay because the Surveillance can only be performed during recirculation loop operation, and the 4 hour period provides a reasonable time period in which to establish conditions appropriate for data collection and evaluation. Currently, the Surveillance is required whenever there is recirculated flow and the reactor is in the Startup/Hot Standby or Run MODES. The second Note permits a delay in performing the Surveillance until the plant exceeds 25% RTP, because during low flow conditions, jet pump noise approaches the threshold response of the flow instrumentation, which precludes collection of repeatable and meaningful data. This change is consistent with NUREG-1433, Revision 1.

L2 CTS 4.6.G.3 requires that individual jet pump differential pressure not vary from the average of all jet pump differential pressures by more than 10%. ITS SR 3.4.2.1 requires that the differential pressure variation from established patterns be not more than 20%. This is a relaxation of requirements and is less restrictive. This change is acceptable because it is consistent with the recommendations provided in General Electric Service Information Letter (SIL) No. 330, Jet Pump Beam Cracks," and NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure." SIL-330 recommends the 10% criteria be used for plants designed with individual jet pump flow indicators. When measured by jet pump diffuser-to-lower plenum differential pressure, the equivalent criteria is 20% due to the relationship between flow and differential pressure. Since JAFNPP utilizes jet pump differential pressures measurement, the variance allowed should have been 20% as was recommended in SIL-330 and NUREG/CR-3052 and currently used in

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

CTS 4.6.G.A.b.2 for single loop operation. The proposed criteria are acceptable since they are consistent with the recommendations in SIL-330 and NUREG/CR-3052.

TECHNICAL CHANGES - RELOCATIONS

None

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NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.2 - JET PUMPS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change adds two Notes to the SR. One relaxes the Surveillance Frequency to allow a 4 hour delay in completion of the performance of the Surveillance after the associated recirculation loop is in operation, and the other allows a delay in completion of the performance of the Surveillance until 24 hours after the plant exceeds 25% RTP. The proposed change does not increase the probability of an accident. Jet pumps are not assumed to be initiators of any analyzed event. The Notes allow time after the loop is placed in operation to establish appropriate conditions for the Surveillance to be performed. The Surveillance is not required to be performed at power levels less than 25% because during low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of meaningful data. The proposed change provides confirmation of the Operability of the jet pumps at the earliest opportunity when the jet pumps are required. In addition, the most common outcome of the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. As a result, the consequences of an accident are not affected by this change. This change will not alter assumptions relative to the mitigation of an accident or transient event. 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?

   The proposed changes to the Frequency will not create the possibility of an accident. The Surveillance Requirement is being performed to confirm the Operability of the jet pumps at the earliest opportunity where meaningful data can be collected when the jet pumps are required. This change will not physically alter the plant (no new or different type of

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

2. (continued)

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?

This change adds two Notes to the Surveillance which relax the Surveillance Frequency to allow a 4 hour delay in completion of the performance of the Surveillance after the associated recirculation loop is in operation, and to not require the completion of the performance of the Surveillance until 24 hours after the plant exceeds 25% RTP. The margin of safety is not significantly reduced because the proposed changes to the Surveillance Frequency will continue to provide the necessary assurance of Operability of the jet pumps at the earliest opportunity. These changes effectively extend the initial performance of the Surveillance Requirement by 4 or 24 hours. This is considered acceptable since the most common outcome to the performance of a Surveillance is the successful demonstration that the acceptance criteria are satisfied. In addition, these changes provide the benefit of allowing the Surveillance to be postponed until plant conditions exist where the Surveillance can be performed. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.2 - JET PUMPS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change revises the allowable jet pump differential pressure variation of ≤ 10% from the average of all jet pump differential pressures, to a variation of ≤ 20% from established patterns, which, in effect, provides a larger window of acceptable jet pump performance. The proposed change is consistent with the recommendations provided in General Electric Service Information Letter (SIL) No. 330, Jet Pump Beam Cracks," and NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure." SIL-330 recommends the 10% criteria be used for plants designed with individual jet pump flow indicators. When measured by jet pump diffuser-to-lower plenum differential pressure, the equivalent criteria is 20% due to the relationship between flow and differential pressure. Since JAFNPP does not have individual jet pump flow indicators and utilizes the diffuser-to-lower plenum differential pressure measurement, the range allowed should be 20%, as recommended in SIL-330 and NUREG/CR-3052. The proposed change does not increase the probability of an accident. Jet pumps are not assumed to be an initiator of any analyzed event. The proposed change does not alter assumptions relative to the mitigation of an accident or transient. As a result, the proposed change does not affect the consequences of an accident. Therefore, this change 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 involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change revises the allowable jet pump differential pressure variation of ≤ 10% from the average of all jet pump differential pressures, to a variation of ≤ 20% from established patterns, which, in effect, provides a larger

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

2. (continued)

window of acceptable jet pump performance. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The proposed change adjusts the jet pump Surveillance acceptance criteria from 10% to 20% for individual jet pump diffuser-to-lower plenum differential pressure variations from the established pattern. This change corrects an error in the Technical Specifications. The error resulted in the JAFNPP acceptance criteria being more conservative than the criteria contained in SIL-330 and NUREG/CR-3052. The margin of safety is not significantly reduced because the proposed changes to the acceptance criteria will continue to verify jet pump Operability. The changes reflect the recommendations in SIL-330 and NUREG/CR-3052. The safety analysis assumptions will still be maintained, thus no question of safety exists. In addition, this change provides the benefit of avoiding a shutdown transient when the jet pumps are still capable of performing their safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

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MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Jet Pumps

[3.6.G] LCO 3.4.2 All jet pumps shall be OPERABLE.

[3.6.G] APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.6.G] A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

CTS

[4.6.G]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 ----------NOTES---------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. ----------	
Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop:	24 hours DBI
a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns. (recirculation loop)	PA2
b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.	
c. Each jet pump flow differs by ≤ 10% from established patterns.	DBI

Reviewer's Note: An acceptable option to these criteria for jet pump OPERABILITY can be found in the BWR/6 ITS, NUREG-1434. PA1

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RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The Reviewer's type Note has been deleted since it was not intended to be maintained in the plant specific ITS.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 SR 3.4.2.1.c is deleted because JAFNPP does not have individual jet pump flow instrumentation; and therefore, this criterion is not needed.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

None

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MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 Jet Pumps

BASES

BACKGROUND

The Reactor ~~Coolant~~ Water Recirculation System is described in the Background section of the Bases for LCO 3.4.1, "Recirculation Loops Operating," which discusses the operating characteristics of the system and how these characteristics affect the Design Basis Accident (DBA) analyses. PA1

PA2

the reactor vessel internals, and in conjunction with

The jet pumps are part of the Reactor ~~Coolant~~ Water Recirculation System ~~and~~ are designed to provide forced circulation through the core to remove heat from the fuel. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Because the jet pump suction elevation is at two-thirds core height, the vessel can be reflooded and coolant level maintained at two-thirds core height even with the complete break of ~~the~~ a recirculation loop pipe that is located below the jet pump suction elevation. PA2

PA3

Each reactor coolant recirculation loop contains ~~ten~~ 10 jet pumps. Recirculated coolant passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the drive flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

APPLICABLE SAFETY ANALYSES

Jet pump OPERABILITY is an ~~explicit~~ implicit assumption in the design basis loss of coolant accident (LOCA) analysis evaluated in Reference 1. PA2

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If the structural system, including the beam holding a jet pump in place, fails, jet pump displacement and performance degradation could occur, resulting in an increased flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

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

Jet pumps satisfy Criterion 2 of ~~the NRC Policy Statement.~~ X1

LCO

The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two-thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure that operation of the Reactor ~~Coolant~~ Water Recirculation System will be consistent with the assumptions used in the licensing basis analysis (Ref. 1). PA1

APPLICABILITY

PA2 "Recirculation Loops Operating"

In MODES 1 and 2, the jet pumps are required to be OPERABLE since there is a large amount of energy in the reactor core and since the limiting DBAs are assumed to occur in these MODES. This is consistent with the requirements for operation of the Reactor ~~Coolant~~ Water Recirculation System (LCO 3.4.1). PA1

In MODES 3, 4, and 5, the Reactor ~~Coolant~~ Water Recirculation System is not required to be in operation, and when not in operation, sufficient flow is not available to evaluate jet pump OPERABILITY.

ACTIONS

A.1

An inoperable jet pump can increase the blowdown area and reduce the capability ~~of~~ to reflood~~ing~~ during a design basis LOCA. If one or more of the jet pumps are inoperable, 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 Completion Time of 12 hours is PA2

(continued)

BASES

ACTIONS

A.1 (continued)

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 3.4.2.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. ~~8~~ 3). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. ~~2 and 3~~ 3 and 4). Each recirculation loop must satisfy one of the performance criteria provided. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation ~~loop~~ pump flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns", engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow and ~~loop~~ recirculation flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship may indicate a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the recirculation pump

(continued)

X2

Insert SR 3.4.2.1-1

Jet Pump OPERABILITY is considered acceptable prior to startup of the plant following a refueling outage due to acceptable results obtained during the previous operating cycle, or by visual inspection of the jet pumps.

Insert SR 3.4.2.1-2 X2

An inoperable jet pump may, in the event of a design basis accident, increase the blowdown area and reduce the capability to reflood the core. Thus, the requirement for shutdown of the plant exists with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance for degradation on a prescribed schedule. During single loop operation (SLO), the jet pump OPERABILITY surveillance is only performed for the jet pumps in the operating recirculation loop, as the loads on the jet pumps in the inactive loop have been demonstrated through operating experience at other BWRs to be very low due to the low flow in the reverse direction through them. The jet pumps in the non-operating recirculation loop during SLO are considered OPERABLE based on this low expected loading, acceptable surveillance results obtained during two recirculation loop operation prior to entering SLO, or by visual inspection of the jet pumps during outages. Upon startup of an idle recirculation loop when THERMAL POWER is greater than 25% of RATED THERMAL POWER, the specified jet pump surveillances are required to be performed for the previously idle loop within 4 hours, as specified in the SR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

flow and loop flow versus pump speed relationship must be verified.

Individual jet pumps in a recirculation loop normally do not have the same flow. The unequal flow is due to the drive flow manifold, which does not distribute flow equally to all risers. The flow or jet pump diffuser to lower plenum differential pressure pattern or relationship of one jet pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps. This may be indicated by an increase in the relative flow for a jet pump that has experienced beam cracks.

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 3). Normal flow ranges and established jet pump flow and differential pressure patterns are established by plotting historical data as discussed in Reference 2.

The 24 hour Frequency has been shown by operating experience to be timely for detecting jet pump degradation and is consistent with the Surveillance Frequency for recirculation loop OPERABILITY verification.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

Note 2 allows this SR not to be performed when THERMAL POWER is ≤ 25% of RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 16.3.
2. GE Service Information Letter No. 330, June 9, 1990.
3. NUREG/CR-3052, November 1984.

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

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.2 - JET PUMPS

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Bases are modified to reflect plant specific nomenclature.

PA2 Editorial changes have been made for clarification, correction, or improvement with no change in intent.

PA3 Bases are modified to maintain consistency with the Writer's Guide for the Restructured Technical Specifications.

PA4 Bases are modified to reflect changes made to the Specifications.

PA5 The statement in the Bases for SR 3.4.2.1 has been deleted because it is misleading. An increase in flow could be indicative of other problems.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 The brackets have been removed and the proper plant specific Reference has been included.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN 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. Subsequent References have been renumbered as required.

X2 Additional discussion added to Bases to address OPERABILITY of jet pumps in an idle recirculation loop during single loop operation and during plant startup following a refueling outage.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.2

Jet Pumps

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Jet Pumps

LCO 3.4.2 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	----------NOTES---------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. ----------	
	Verify at least one of the following criteria (a or b) is satisfied for each operating recirculation loop: a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and recirculation loop jet pump flow to recirculation pump speed ratio differs by ≤ 5% from established patterns. b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.	24 hours

RAI 34-GEN

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 Jet Pumps

BASES

BACKGROUND

The Reactor Water Recirculation System is described in the Background section of the Bases for LCO 3.4.1, "Recirculation Loops Operating," which discusses the operating characteristics of the system and how these characteristics affect the Design Basis Accident (DBA) analyses.

The jet pumps are part of the reactor vessel internals, and in conjunction with the Reactor Water Recirculation System are designed to provide forced circulation through the core to remove heat from the fuel. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Because the jet pump suction elevation is at two-thirds core height, the vessel can be reflooded and coolant level maintained at two-thirds core height even with the complete break of a recirculation loop pipe that is located below the jet pump suction elevation.

Each reactor coolant recirculation loop contains 10 jet pumps. Recirculated coolant passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the drive flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

APPLICABLE SAFETY ANALYSES

Jet pump OPERABILITY is an implicit assumption in the design basis loss of coolant accident (LOCA) analysis evaluated in Reference 1.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If the structural system, including the beam holding a jet pump in place, fails, jet pump displacement and performance degradation could occur, resulting in an increased flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

Jet pumps satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two-thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure that operation of the Reactor Water Recirculation System will be consistent with the assumptions used in the licensing basis analysis (Ref. 1).

APPLICABILITY

In MODES 1 and 2, the jet pumps are required to be OPERABLE since there is a large amount of energy in the reactor core and since the limiting DBAs are assumed to occur in these MODES. This is consistent with the requirements for operation of the Reactor Water Recirculation System (LCO 3.4.1, "Recirculation Loops Operating").

In MODES 3, 4, and 5, the Reactor Water Recirculation System is not required to be in operation, and when not in operation, sufficient flow is not available to evaluate jet pump OPERABILITY.

ACTIONS

A.1

An inoperable jet pump can increase the blowdown area and reduce the capability to reflood during a Design Basis LOCA. If one or more of the jet pumps are inoperable, the plant must be brought to a MODE in which the LCO does not apply.

(continued)

BASES

ACTIONS

A.1 (continued)

To achieve this status, the plant must be brought to MODE 3 within 12 hours. The 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 3.4.2.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 3). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 3 and 4). Each recirculation loop must satisfy one of the performance criteria provided. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, recirculation loop jet pump flow, and recirculation pump flow, these relationships may need to be re-established each cycle. Jet Pump OPERABILITY is considered acceptable prior to startup of the plant following a refueling outage due to acceptable results obtained during the previous operating cycle, or by visual inspection of the jet pumps. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns", engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

RAI 3.4-GEN

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

An inoperable jet pump may, in the event of a design basis accident, increase the blowdown area and reduce the capability to reflood the core. Thus, the requirement for shutdown of the plant exists with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance for degradation on a prescribed schedule. During single loop operation (SLO), the jet pump OPERABILITY surveillance is only performed for the jet pumps in the operating recirculation loop, as the loads on the jet pumps in the inactive loop have been demonstrated through operating experience at other BWRs to be very low due to the low flow in the reverse direction through them. The jet pumps in the non-operating recirculation loop during SLO are considered OPERABLE based on this low expected loading, acceptable surveillance results obtained during two recirculation loop operation prior to entering SLO, or by visual inspection of the jet pumps during outages. Upon startup of an idle recirculation loop when THERMAL POWER is greater than 25% of RATED THERMAL POWER, the specified jet pump surveillances are required to be performed for the previously idle loop within 4 hours, as specified in the SR.

The recirculation pump speed operating characteristics (recirculation pump flow and recirculation loop jet pump flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship may indicate a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the recirculation pump flow and recirculation loop jet pump flow versus pump speed relationship must be verified.

Individual jet pumps in a recirculation loop normally do not have the same flow. The unequal flow is due to the drive flow manifold, which does not distribute flow equally to all risers. The jet pump diffuser to lower plenum differential pressure pattern or relationship of one jet pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 3). Normal flow ranges and established jet pump differential pressure patterns are established by plotting historical data as discussed in Reference 3.

The 24 hour Frequency has been shown by operating experience to be timely for detecting jet pump degradation and is consistent with the Surveillance Frequency for recirculation loop OPERABILITY verification.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds 25% of RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.

REFERENCES

1. UFSAR, Section 14.6.

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

3. GE Service Information Letter No. 330, including Supplement 1, Jet Pump Beam Cracks, June 9, 1990.

4. NUREG/CR-3052, Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure, November 1984.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

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

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.4.3

A1

JAFNPP

[3.4.3]

3.6 (cont'd)

D. Safety/Relief Valves

Applicability

A2

[LCO 3.4.3]

1. During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212 F, the safety mode of at least 9 of 11 safety/relief valves shall be operable. The Automatic Depressurization System valves shall be operable as required by specification 3.5.D.

MODES 1, 2 and 3

A2

A3

[3.4.3]

4.6 (cont'd)

E. Safety/Relief Valves

1. At least 5 of the 11 safety/relief valves shall be bench checked or replaced with bench checked valves every 24 months. ~~All valves shall be tested every 48 months.~~ The testing shall demonstrate that each valve tested actuates at 1145 psig ±3%. Following testing, lift settings shall be 1145 psig ±1%.

[SR 3.4.3.1]

In accordance with IST program

LA3

Page 1 of 3

Specification 3.4.3

A1

1.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to limits on reactor coolant system pressure.

OBJECTIVE:

To establish a limit below which the integrity of the Reactor Coolant System is not threatened due to an overpressure condition.

SPECIFICATION:

1. The reactor vessel dome pressure shall not exceed 1,325 psig at any time when irradiated fuel is present in the reactor vessel.

ITS: Chapter 2.0

2.2 REACTOR COOLANT SYSTEM

APPLICABILITY:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor coolant system safety limits from being exceeded.

OBJECTIVE:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

SPECIFICATION:

1. The Limiting Safety System setting shall be specified below:

   A. Reactor coolant high pressure scram shall be ≤1,080 psig.

   B. At least 9 of the 11 reactor coolant system safety/relief valves shall have a nominal setting of 1,145 psig with an allowable setpoint error of ±3 percent.

[SR 3.4.3.1]

See ITS: 3.3.1.1

Specification 3.4.3

A1

~~3.6 (cont'd)~~

M1

Be in MODE 3 in 12 hours

[ACTION A]

2. If Specification 3.6.E.1 is not met, the reactor shall be placed in a cold condition within ~~24~~ 36 hours. L1

3. ~~Low power physics testing and reactor operator training shall~~ be permitted with inoperable components as specified in Specification 3.6.E.1 above, provided that reactor coolant temperature is ≤212 °F and the reactor vessel is vented or the reactor vessel head is removed.

4. The provisions of Specification 3.0.D are not applicable.

<See ITS 3.10.8>

~~4.6 (cont'd)~~

LA1

2. At least one safety/relief valve shall be disassembled and inspected every 24 months.

3. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

<See ITS 3.5.1>

required A4

LA2

4. Manually open each safety/relief valve while bypassing steam to the condenser and observe a ≥10% closure of the turbine bypass valves, to verify that the safety/relief valve has opened. This test shall be performed at least every 24 months while in the RUN mode and within the first 12 hours after steam pressure and flow are adequate to perform the test.

LA2

[SR 3.4.3.2]

[Note to SR 3.4.3.2]

M2

<Add: STAGGERED TEST BASIS for each solenoid>

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JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.4.3 - SAFETY/RELIEF VALVES (S/RVs)

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (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 3.6.E.1 Applicability is "during reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212°F." ITS 3.4.3 Applicability is in MODES 1, 2, and 3. The CTS Applicability of "during reactor power operating conditions," and "whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212°F," are encompassed by the ITS MODES of applicability. The CTS Applicability, "prior to startup from a cold condition," is consistent with CTS 3.0.D and ITS LCO 3.0.4, which require that an LCO be met prior to entry into the MODE or other specified condition in the Applicability. Since no technical requirements are altered, this change is administrative. This change is consistent with NUREG-1433, Revision 1.

A3 CTS 3.6.E.1 specifies that the Automatic Depressurization System (ADS) valves shall be OPERABLE as required by CTS 3.5.D. This statement reminds the reader that another Specification is also Applicable, and is not retained in the ITS. Since no technical requirements are altered, this change is administrative. This change is consistent with NUREG-1433, Revision 1.

A4 CTS 4.6.E.4 is revised to reflect that only each "required" S/RV need be manually opened. Since CTS 3.6.E.1 states that 9 of 11 S/RVs are required to be OPERABLE, and the Technical Specifications only apply to "required" equipment, this change is considered administrative. This change is consistent with NUREG-1433, Revision 1.

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A5 Not used.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.6.E.2 requires the reactor to be placed in a cold condition within 24 hours if the requirements of CTS 3.6.E.1 can not be met (less than the minimum number of Operable safety/relief valves). In ITS 3.4.3 this condition is addressed in ITS 3.4.3 ACTION A. ITS 3.4.3 ACTION A requires the plant to be in MODE 3 in 12 hours (Required Action A.1). In addition the time to reach cold condition (MODE 4) has been extended to 36 hours (see L1). The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. However, the 12 hour Completion Time ensures timely action is taken to place the plant in a shutdown condition (MODE 3). The consequences of an overpressurization event is significantly reduced when the plant is shutdown. This change is consistent with NUREG-1433, Revision 1.

M2 CTS 4.6.E.4 requires the safety/relief valves to be manual opened every 24 months. ITS SR 3.4.3.2 requires this same manual opening but requires the actuation to be initiated on a Staggered Test Basis for each valve solenoid. This will ensure that a different solenoid will be used to actuate the valve every 24 months and is considered more restrictive since the current requirement does not specify which solenoid to use. This change is consistent with NUREG-1433, Revision 1 and is necessary to ensure both solenoids are Operable.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The requirement in CTS 4.6.E.2 to disassemble and inspect one safety/relief valve every 24 months is proposed to be relocated to the JAFNPP UFSAR because it is a maintenance related activity that does not directly relate to S/RV Operability. This inspection is a preventative maintenance type requirement. The failure to perform this requirement does not necessarily result in an inoperable S/RV. This requirement is oriented toward long term S/RV Operability and does not have an immediate impact on S/RV Operability. S/RV Operability is verified by the SRs maintained in ITS 3.4.3. In addition, procedural controls on S/RV inspections are sufficient to ensure that the S/RV receives the necessary inspections. As a result, this requirement is not necessary to be included in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA2 The methods in CTS 4.6.E.4 for verifying the safety/relief valves has opened (i.e., while bypassing steam to the condenser, etc) and the detail that the test must be performed in Run are proposed to be relocated to the Bases. These details are not necessary to ensure Operability of the S/RVs. The requirements of ITS 3.4.3 and the associated SRs are adequate to ensure that the S/RVs are maintained Operable. SR 3.4.3.2 will require each required S/RV to be manually actuated after reactor steam dome pressure and flow are adequate to perform this test. The Bases for this SR will prescribe the test method and the conditions for performing the test. In addition, the Bases discusses that the pressure and flow conditions will require the plant to be in MODE 1, which has been shown to be an acceptable condition to perform this test. This test will cause a small neutron flux transient which may cause a scram while operating close to the Average Power Range Monitors Neutron Flux-High (Startup) Allowable Value in MODE 2. As such these methods of verification and details that the plant must be in Run are not necessary to be included 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.

LA3 The requirement in CTS 4.6.E.1 that at least 5 of the 11 S/RVs be bench checked or replaced with bench checked valves every 24 months; and that all valves be tested every 48 months are proposed to be relocated to the Inservice Testing Program. The Frequency is revised in ITS SR 3.4.3.1 to, "In accordance with the Inservice Testing (IST) Program". The requirement in ITS SR 3.4.3.1 to verify the lift setpoints of the required S/RVs in accordance with the Inservice Testing Program is adequate to ensure the valves are OPERABLE. Testing of pumps and 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 this detail is not necessary to be included in the ITS to provide adequate protection of the public health and safety. Changes to the testing Frequency in the IST Program will be controlled by the provisions of 10 CFR 50.59.

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LA4 Not used.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.6.E.2 requires the reactor to be placed in a cold condition within 24 hours if the requirements of CTS 3.6.E.1 cannot be met (less than the minimum number of Operable safety/relief valves). In ITS 3.4.3, this condition is addressed in ITS 3.4.3 ACTION A. The proposed requirement, ITS 3.4.3, Required Action A.2, extends the time allowed for the plant to reduce temperature to be in MODE 4, from 24 hours to 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The consequences of an accident are not significantly increased because ITS 3.4.3 Required Action A.1 will require the plant be placed in MODE 3 within 12 hours. This change reduces the time the reactor would be allowed to continue to operate once the condition is identified. The consequences of a pressurization event is significantly reduced when the reactor is shutdown and a controlled cooldown is already in progress. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.3 - SAFETY/RELIEF VALVES (S/RVs)

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change extends the time allowed for the plant to achieve Cold Shutdown conditions from 24 hours to 36 hours with one or more required S/RVs inoperable. Shutdown Completion Times are not assumed in the initiation of any analyzed event. The change will not allow continuous operation with one or more required S/RVs inoperable. In addition, the consequences of an accident are not increased because LCO 3.4.3 Required Action A.1 will require that the plant be placed in MODE 3 within 12 hours once the determination is made that the LCO is not met. This change reduces the time the reactor would be allowed to continue to operate once the condition is identified. The consequences of an overpressurization event are significantly reduced when the reactor is shutdown and a controlled cooldown is already in progress. In addition, the consequences of an event occurring during the proposed shutdown Completion Time are the same as the consequences of an event occurring during the existing shutdown Completion Time. Therefore, the change does not involve a significant increase in the probability or consequences of an event 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 any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change extends the Completion Time for reaching MODE 4 from 24 hours to 36 hours. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The proposed change extends the time allowed for the plant to achieve Cold Shutdown conditions from 24 hours to 36 hours with one or more required S/RVs inoperable. There is no significant reduction in the margin of safety because ITS 3.4.3 Required Action A.1 will require that

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

3. (continued)

the plant be placed in MODE 3 within 12 hours once the determination is made that the requirements of the LCO are not met. This concurrent change reduces the time the reactor would be allowed to continue to operate once the condition is identified. The consequences of an overpressurization event are significantly reduced when the reactor is shutdown and a controlled cooldown is already in progress. In addition, this change provides the benefit of a reduced potential for a plant event that could challenge safety systems by providing additional time to reduce pressure in a controlled and orderly manner. Therefore, this change does not involve a significant reduction in a margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (S/RVs)

CLB1

[3.6.E.1] LCO 3.4.3 The safety function of ~~NN~~ 9 S/RVs shall be OPERABLE.

[3.6.E.1] APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One [or two][required] S/RV[s] inoperable.~~	~~A.1 Restore the [required] S/RV[s] to OPERABLE status.~~	~~14 days~~
~~B. Required Action and associated Completion Time of Condition A not met.~~ ~~OR~~ [3.6.E.2] A. ~~[Three]~~ One or more ~~[required]~~ S/RVs inoperable.	A ~~B.1~~ Be in MODE 3.	12 hours
	~~AND~~ A ~~B.2~~ Be in MODE 4.	36 hours

CLB2

CLB2

SURVEILLANCE REQUIREMENTS

PA1

	SURVEILLANCE	FREQUENCY
[4.6.E.1] [2.2.1.B] SR 3.4.3.1	Verify the safety function lift setpoints of the required S/RVs ~~are as follows:~~ ~~Number of S/RVs~~ ~~Setpoint (psig)~~ ~~[4] [1090 ± 32.7]~~ ~~[4] [1100 ± 33.0]~~ ~~[3] [1110 ± 33.3]~~ is 1145 ± 34.3 psig.5 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program ~~or [18] months~~ X1 CLB3
[4.6.E.4] [M2] SR 3.4.3.2	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify each required S/RV opens when manually actuated. PA1	~~[18]~~ 24 months on a STAGGERED TEST BASIS for each valve solenoid X2

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

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

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.3 - SAFETY/RELIEF VALVES (S/RVs)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The brackets have been removed and the proper plant specific value has been included consistent with CTS 3.6.E.1.

CBL2 The bracketed ACTION has been deleted since it does not apply to JAFNPP. The LCO contains the required number of S/RVs to satisfy the overpressurization safety analysis. If one of the required S/RVs are inoperable a shutdown must commence. All subsequent CONDITIONS and Required Actions have been renumbered, where applicable.

CBL3 The bracketed requirements have been revised in SR 3.4.3.1 consistent with the requirements in CTS 4.6.E.1. All required S/RVs will have the same safety function lift setpoint. The specified value is consistent with CTS 4.6.E.1.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The word "required" has been included since all S/RVs are not required to be Operable.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 The brackets have been removed and the proper plant specific Frequency included in SR 3.4.3.1 consistent with CTS 4.6.E.1 as modified by LA3.

X2 The S/RV manual actuation test is currently required to be performed on a 24 month Frequency (CTS 4.6.E.4). The requirement to test each valve on a STAGGERED TEST BASIS for each valve solenoid has been added in accordance with M2.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

(Ref. 1) DB1

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

PA1

However for the purposes of this LCO, only the safety mode is required.

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

PA1

Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."

DB2

INSERT BKGD

DB2

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, 11 S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design

DB1

5

3 and 4

CLB1

9

(Ref. 4)

DB1

9 S/RVs are — DB2

(continued)

DB2

Insert BKGD

All S/RVs can be opened manually in the relief mode from the control room by its associated two-position switch. If one of these switches is placed in the open position the logic output will energize the associated S/RV solenoid control valve directing the pneumatic supply to open the valve. Seven of these S/RV solenoid control valves can also be energized by the relay logic associated with the Automatic Depressurization System (ADS). ADS requirements are specified in LCO 3.5.1, "ECCS-Operating." In addition each S/RV can be manually operated from another control switch located at the ADS auxiliary panel located outside the control room. These switches will energize a different S/RV solenoid control valve. The details of S/RVs pneumatic supply and mechanical operation in the relief mode are described in Reference 2.

BASES

APPLICABLE SAFETY ANALYSES (continued)

pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 3 discusses additional events that are expected to actuate the S/RVs.

S/RVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The safety function of N S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1 and 2). The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The S/RV setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the FSAR are based on these setpoints, but also include the additional uncertainties of ±3% of the nominal setpoint drift to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, N S/RVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

(continued)

BASES

APPLICABILITY (continued)

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

A.1

With the safety function of one [or two] [required] S/RV[s] inoperable, the remaining OPERABLE S/RVs are capable of providing the necessary overpressure protection. Because of additional design margin, the ASME Code limits for the RCPB can also be satisfied with two S/RVs inoperable. However, the overall reliability of the pressure relief system is reduced because additional failures in the remaining OPERABLE S/RVs could result in failure to adequately relieve pressure during a limiting event. For this reason, continued operation is permitted for a limited time only.

The 14 day Completion Time to restore the inoperable required S/RVs to OPERABLE status is based on the relief capability of the remaining S/RVs, the low probability of an event requiring S/RV actuation, and a reasonable time to complete the Required Action.

CLB2

CLB2 A

B.1 and B.2

CLB2

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of the inoperable required S/RVs cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1, or if the safety function of [three] or more [required] S/RVs is inoperable, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 4. The demonstration of the S/RV safe lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

The 18 month Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings.

SR 3.4.3.2

A manual actuation of each required S/RV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is 1920 psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by at least 1.25 turbine bypass valves open, or total steam flow ≥ 10^6 lb/hr. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable

(continued)

PA2 PA1 s 3 and 4 DB1 X2 PA5 X2 CLB3 while bypassing main steam flow to the condenser and observing ≥ 10 % closure of the turbine bypass valves also CLB3 PA2 DB3 consistent with vendor recommendations 970 two or more DB3 PA1 steam and flow are DB3 These conditions will require the plant to be in MODE 1, which has been shown to be an acceptable condition to perform this test. This test causes a small neutron flux transient which may cause a scram in MODE 2 while operating close to the Average Power Range Monitors Neutron Flux - High (Startup) Allowable Value.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The 18 month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 18 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 3). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Section 5.2.2.2.4.
3. UFSAR, Section 15.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. 10 CFR 50.36 (c)(2)(ii)

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

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

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.3 - SAFETY/RELIEF VALVES (S/RVs)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The brackets have been removed and the proper plant specific value has been included consistent with CTS 3.6.E.1 and the design analysis. Changes have been made to reflect the proper numbers of S/RVs throughout the Bases.

CLB2 The bracketed ACTION has been deleted since it does not apply to JAFNPP. The LCO contains the required number of S/RVs to satisfy the overpressurizaton safety analysis. If one of the required S/RVs are inoperable a shutdown must commence. All subsequent CONDITIONS and Required Actions have been renumbered, where applicable.

CLB3 The Bases for SR 3.4.3.2 have been revised to reflect the current method to demonstrate that an S/RV has opened.

CLB4 The bracketed requirement of +/- 3% is retained in SR 3.4.3.1 consistent with the requirements in CTS 4.6.E.1.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Editorial changes have been made for clarification, correction, or improvement with no change in intent.

PA2 The word "required" has been retained since all S/RVs are not required to be Operable.

PA3 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 References and their associated numbering have been revised to reflect JAFNPP specific information.

DB2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific design or analysis.

DB3 The brackets have been removed and the proper plant testing conditions included in the Bases for SR 3.4.3.2. In addition, the Bases has been revised to reflect the proper justification for these conditions.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.3 - SAFETY/RELIEF VALVES (S/RVs)

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN 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.

X2 The brackets have been removed and the Frequency of in accordance with the Inservice Testing Program retained consistent with CTS 4.6.E.1 as modified by LA3. Changes have been made to the Bases to reflect this proposed Frequency.

X3 The S/RV manually actuation test (SR 3.4.3.2) is currently required to be performed on a 24 month Frequency (CTS 4.6.E.4). The requirement to test each valve on a STAGGERED TEST BASIS for each solenoid valve has been added in accordance with M2. The Bases have been modified accordingly.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.3

Safety/Relief Valves (S/RVs)

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (S/RVs)

LCO 3.4.3 The safety function of 9 S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3.	12 hours
	AND	
	A.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoint of the required S/RVs is 1145 ± 34.3 psig. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ----------	
	Verify each required S/RV opens when manually actuated.	24 months on a STAGGERED TEST BASIS for each valve solenoid

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code (Ref. 1) requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The S/RVs can actuate by either of two modes: the safety mode or the relief mode. However, for the purposes of this LCO, only the safety mode is required. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

All S/RVs can be opened manually in the relief mode from the control room by its associated two-position switch. If one of these switches is placed in the open position the logic output will energize the associated S/RV solenoid control valve directing the pneumatic supply to open the valve. Seven of these S/RV solenoid control valves can also be energized by the relay logic associated with the Automatic Depressurization System (ADS). ADS requirements are specified in LCO 3.5.1, "ECCS-Operating." In addition each S/RV can be manually operated from another control switch located at the ADS auxiliary panel located outside the control room. These switches will energize a different S/RV solenoid control valve. The details of S/RVs pneumatic supply and mechanical operation in the relief mode are described in Reference 2.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Refs. 3 and 4). For the purpose of the analyses (Ref. 4), 9 S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that 9 S/RVs are capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig (at the vessel bottom) is met during the most severe pressurization transient.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 4 discusses additional events that are expected to actuate the S/RVs.

S/RVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

The safety function of 9 S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 3 and 4). The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The single nominal S/RV setpoint is established (Ref. 3) to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The single nominal S/RV setpoint is set below the RPV design pressure (1250 psig) in accordance with ASME Code requirements. The transient evaluations in Reference 4 are based on this single setpoint, but also includes the additional uncertainties of ± 3% of the nominal setpoint to provide an added degree of conservatism.

(continued)

BASES

LCO (continued)

Operation with fewer valves OPERABLE than specified, or with setpoints outside the analysis limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, nine S/RVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of the inoperable required S/RVs cannot be restored to OPERABLE status, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the required S/RVs open at the pressures assumed in the safety analysis of References 3 and 4. The demonstration of the S/RV safe lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

SR 3.4.3.2

A manual actuation of each required S/RV is performed while bypassing main steam flow to the condenser and observing ≥ 10% closure of the turbine bypass valves to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can also be demonstrated by the response of the turbine control valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is 970 psig (the pressure consistent with vendor recommendations). Adequate steam flow is represented by two or more turbine bypass valves open, or total steam flow ≥ 10^6 lb/hr. These conditions will require the plant to be in MODE 1, which has been shown to be an acceptable condition to perform this test. This test causes a small neutron flux transient which may cause a scram in MODE 2 while operating close to the Average Power Range Monitors Neutron Flux-High (Startup) Allowable Value. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required steam pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The 24 month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 24 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 6). Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Section 4.4.
3. UFSAR, Section 14.5.1.2.
4. UFSAR, Section 16.9.3.2.3.
5. 10 CFR 50.36(c)(2)(ii).
6. ASME, Boiler and Pressure Vessel Code, Section XI.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.4

RCS Operational LEAKAGE

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

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.4

RCS Operational LEAKAGE

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.4.4

A1

RCS OPERATIONAL

[3.4.4] ~~B.~~ Coolant Leakage

MODES 1, 2 and 3 M3

[Applicability] 1. Anytime irradiated fuel is in the reactor vessel and the reactor coolant temperature is above 212°F, the reactor coolant leakage into the primary containment shall be limited to:

[LCO 3.4.4]

[LCO 3.4.4.b] a. 5 gpm unidentified leakage

no pressure boundary leakage M1

[LCO 3.4.4.d] b. 2 gpm increase in unidentified leakage within any 24 hour period. (This limitation shall apply only after a period of 24 hours at operating pressure.)

in MODE 1 L1

[LCO 3.4.4.c] c. The total reactor coolant leakage into the primary containment shall not exceed 25 gpm.

averaged over the previous 24 hr period L2

RCS OPERATIONAL

[3.4.4] ~~D.~~ Coolant Leakage

A2

[SR 3.4.4.1] 1. Reactor coolant leakage rate inside the primary containment shall be monitored and recorded once every 4 hours utilizing the Primary Containment Sump Monitoring System (equipment drain sump monitoring and floor drain sump monitoring).

L A1

3.6 (cont'd)

[ACTION A] 2. With reactor coolant system leakage greater than the limits specified in 3.6.D.1.a or 3.6.D.1.c, the leakage rate shall be reduced to within these limits within 4 hours or

[ACTION C] the reactor shall be in at least the hot standby condition within the following 12 hours and in cold condition within the next 24 hours.

[ACTION B] 3. With an increase in unidentified reactor coolant system leakage equal to or greater than the limit specified in 3.6.D.1.b, the source of the leakage shall be identified

[Required Action B.2] within 4 hours or the reactor shall be in at least hot standby condition within the next 12 hours and in cold condition within the following 24 hours.

[ACTION C]

4. The Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring and Floor Drain Sump Monitoring) and the Continuous Atmosphere Monitoring System (Gaseous and Particulate) shall be operable when the reactor coolant leakage limits of Specification 3.6.D.1 are in effect.

add 2nd condition in ACTION C

M1

4.6 (cont'd)

2. Not Used

L3

add Required Action B.1

3. Not Used

and verified not to be type 304 or 316 austenitic Stainless steel, or LEAKAGE increase reduced to within limits

M2

4. The Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring and Floor Drain Sump Monitoring) instrumentation shall be calibrated and checked as specified in Surveillance Requirement 4.2.E. Continuous Atmosphere Monitoring System (Gaseous and Particulate) instrumentation shall be functionally tested and calibrated as specified in Table 4.6-2.

see ITS 3.4.5

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.4

RCS Operational LEAKAGE

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.4.4 - RCS OPERATIONAL LEAKAGE

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specifications (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 The requirement to record the results in CTS 4.6.D.1 (ITS SR 3.4.4.1) is proposed to be deleted. This requirement duplicates the requirements of 10 CFR 50 Appendix B, Section XVII (Quality Assurance Records): maintain records of activities affecting quality, including the results of tests (i.e., Technical Specification Surveillances). Compliance with 10 CFR 50 Appendix B is required by the JAFNPP Operating License. The details of the regulations within the Technical Specifications are repetitious and unnecessary. Therefore, retaining the requirement to perform the associated surveillances and eliminating the details from Technical Specifications that are found in 10 CFR 50 Appendix B is considered a presentation preference, which is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.6.D.1 is revised to adopt the requirement that no Reactor Coolant System (RCS) pressure boundary LEAKAGE exist (proposed LCO 3.4.4.a), and should it occur, to be in MODE 3 in 12 hours and in MODE 4 in 36 hours (proposed ACTION C). Since no similar Specification exists, this change imposes additional operational requirements which are more restrictive but necessary to ensure appropriate actions are taken to prevent further degradation of the reactor coolant pressure boundary (RCPB).

M2 CTS 3.6.D.3 requires that the source of an increase in leakage be identified within 4 hours. ITS 3.4.4 Required Action B.2 requires that the source of an increase in LEAKAGE be verified not to be service sensitive type 304 or type 316 austenitic stainless steel within 4 hours. This change seeks to ensure that new or additional RCS LEAKAGE is not the result of intergranular stress corrosion cracking (IGSCC) in the reactor coolant pressure boundary (RCPB). The alternative Required Action is acceptable because the low limit on the rate of increase of unidentified leakage was established as a method of early identification of IGSCC in type 304 and type 316 austenitic stainless steel piping. IGSCC produces tight cracks and the small flow increase limit is capable of providing an early warning of such deterioration. Verification that the source of leakage is not type 304 and type 316 austenitic stainless

TECHNICAL CHANGES - MORE RESTRICTIVE

M2 (continued)

steel eliminates IGSCC as a cause of the leak. This significantly reduces concerns about crack instability and the rapid failure in the RCS boundary. This change imposes additional requirements and is therefore more restrictive but necessary to ensure IGSCC in the RCPB is not the cause of the increased leakage.

M3 CTS 3.6.D requires the reactor coolant leakage into the primary containment to be within limits anytime irradiated fuel is in the reactor vessel and the reactor coolant temperature is above 212°F. ITS 3.4.4 Applicability is during MODES 1, 2 and 3. The ITS Applicability covers additional modes of operation. The CTS requirement to be Operable when the reactor coolant temperature is greater than 212°F only covers ITS MODES 1 and 3. Therefore, the addition of MODE 2 is an additional requirement not explicitly established in the CTS. This added requirement will ensure that if the reactor coolant temperature is below 212°F (MODE 4), reactor coolant leakage is within limits before placing the plant in Startup (MODE 2). This change is considered more restrictive on plant operations since it expands the Applicability requirements but is necessary to ensure that leakage is within limits in those modes of operation where there is a potential for reactor coolant pressure boundary leakage. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 4.6.D.1 details of the methods for performing this surveillance by utilizing the Primary Containment Sump Monitoring System (equipment drain sump monitoring and floor drain sump monitoring) is proposed to be relocated to the Bases. The requirements of proposed SR 3.4.4.1 are adequate for ensuring that RCS leakage is determined to be within the required limits. The details relocated to the Bases are not necessary for ensuring RCS Leakage is determined. As such, these details are not required to be in the ITS to provide adequate protection of 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 The unidentified leakage rate increase limit (CTS 3.6.D.1.b) is proposed to be applicable only in MODE 1 (ITS LCO 3.4.4.d), instead of the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 (continued)

current MODES 1, 2, and 3 (i.e., is at operating pressure after a period of 24 hours). An unidentified LEAKAGE increase of > 2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the LEAKAGE. As the plant starts up and increases pressure, leakage will occur due to the increased pressure. Thus, an increase is detected, and if greater than the limit, could require a plant shutdown, even though there is no safety problem. This proposed change will not require the limit to be applied until MODE 1 is achieved, which is when reactor pressure has effectively stabilized at nominal operating pressure. The overall 5 gpm unidentified Leakage limit will be maintained. This limit is well below the expected flow from a critical sized crack in the primary system.

L2 CTS 3.6.D.1.c requires that total leakage not exceed 25 gpm. ITS 3.4.4 requires that total LEAKAGE not exceed specified limits when averaged over the previous 24 hour period. Total leakage consists of unidentified and identified Leakage. The unidentified Leakage is the more important of the two leakages, and it is being maintained as an instantaneous limit; it is not being averaged to determine unidentified Leakage. The total leakage limit is chosen to ensure the RCS inventory makeup capability and drywell sump capacity is not exceeded. Allowing instantaneous total leakage to be greater than the limit, provided the average total leakage over a 24 hour period is within the limit is acceptable since the current 25 gpm limit is well within the capability of the CRD System pumps and the RCIC System, and is well below the capacity of the drywell equipment drain sump. Additionally, the existing limits associated with unidentified Leakage will still apply.

L3 CTS 3.6.D.3 requires that the source of an increase in the leakage be identified within 4 hours. ITS 3.4.4 Required Action B.1, provides an additional 4 hours to allow the operators to reduce the leakage (or leakage increase) to within acceptable limits before a reactor shutdown must be initiated. This additional 4 hours is acceptable because the leakage limits are significantly below the leakage that would result from a critical sized crack. The critical crack size is indicative of a crack large enough to result in crack instability.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.4

RCS Operational LEAKAGE

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.4 - RCS OPERATIONAL LEAKAGE

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change would revise the Applicability of the unidentified Leakage rate increase to include only MODE 1, instead of the current MODES 1, 2, and 3 (i.e., is at operating pressure after a period of 24 hours). The limit is intended to be applied to changes from normal steady state operational leakage rates. These are typically established at the operating pressures and temperatures consistent with MODE 1. In this manner, a change that indicates a potential problem can be investigated prior to a catastrophic pipe rupture. However, a change during a heatup or startup that does not exceed an unidentified Leakage of 5 gpm, in most cases, does not indicate a potential problem that could result in a catastrophic pipe rupture. The overall unidentified Leakage limit of 5 gpm remains unchanged and will ensure that changes that exceed this limit will not go unrecognized in MODES 2 and 3. Therefore the probability and consequences of a previously analyzed accident are not significantly increased.

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 would revise the Applicability of the unidentified Leakage rate increase to include only MODE 1, instead of the current MODES 1, 2, and 3 (i.e., is at operating pressure after a period of 24 hours). This change does not involve a significant reduction in a margin of safety, because it does not modify the total unidentified

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

3. (continued)

Leakage limit of 5 gpm. This limit is well below the leakage rate expected just prior to the onset of rapid crack propagation. In addition, the proposed change provides the benefit of avoiding a potential plant shutdown transient when no safety concern exists. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.4 - RCS OPERATIONAL LEAKAGE

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change will allow the RCS total Leakage to be averaged over the previous 24 hour period instead of the CTS, which requires the 25 gpm limits to be met at all times. This change has not been identified as an initiator of any accident and does not involve a significant increase in the probability of an accident previously evaluated. The total Leakage limit is not based on any safety analysis limit. It is based on ensuring any Leakage is within the makeup capability of the RCIC and CRD Systems, and the removal capability of the drywell sump pumps. The total Leakage limit consists of the sum of the unidentified and identified Leakage. Since the unidentified Leakage is the more important of the two, it is being maintained in the ITS as an instantaneous limit, and is not being averaged over a 24 hour period. Allowing the instantaneous total Leakage limit to exceed 25 gpm is acceptable as long as the 24 hour average limit is met, since adequate makeup capability will exist and a scheduled Surveillance every 4 hours will identify any large increases. Therefore, this change does 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 involve any design changes, plant modifications, or changes in plant operation. The system will continue to function in the same way as before the change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The proposed change does not change the limit on total RCS Leakage of ≤ 25 gpm. This change will allow the limit to be averaged over the previous 24 hours. This change will not affect the unidentified Leakage limit of ≤ 5 gpm, which is an instantaneous limit. No applicable safety

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

3. (continued)

analysis assumes the total Leakage limit. This limit is based on RCS inventory makeup capability and drywell sump capacity. The change will still maintain total Leakage within the capability of the RCIC and CRD Systems, and the removal capability of the drywell sump pumps. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.4 - RCS OPERATIONAL LEAKAGE

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change will allow an additional 4 hours following the determination that RCS Leakage is exceeding specified limits before a reactor shutdown must be initiated. This additional time is intended to allow the operators to attempt to reduce the leakage rate to within acceptable limits. The probability of an accident is not increased because the amount of time between identification of a leak and the initiation of a reactor shutdown is not considered as an initiator of any accidents previously evaluated. The consequences of an accident will not be increased because the additional 4 hours permitted to investigate and correct the source of RCS Leakage is less than the time it takes for a critical sized crack to develop from these limits. The 5 gpm unidentified Leakage limit is a small fraction of the calculated flow from a critical sized crack in the primary system piping. Exceeding the leakage rate limit does not necessarily violate the absolute unidentified Leakage limit. Crack behavior from experimental programs shows that leakage rates of hundreds of gallons per minute will precede crack instability. The difference between the specified RCS Leakage rate limit and a critical crack leak rate is sufficiently large to allow a time period for corrective action to be taken before the reactor coolant pressure boundary is compromised. 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 will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

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

The proposed change will allow an additional 4 hours following the determination that RCS Leakage is exceeding specified limits before a reactor shutdown must be initiated. This additional time is intended to allow the operators to attempt to reduce the leakage (or leakage increase) to within acceptable limits. RCS Leakage limits are intended to provide early indication of RCS boundary cracks that could be precursors to loss of coolant accidents. The 5 gpm limit is a small fraction of the calculated flow from a critical sized crack in the primary system piping. Crack behavior from experimental programs shows that leakage rates of hundreds of gallons per minute will precede crack instability. The difference between the specified RCS Leakage limits and a critical crack leak rate is sufficiently large to allow a time period for corrective action to be taken before the reactor coolant pressure boundary is compromised. As a result, this change does not affect the current analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.4

RCS Operational LEAKAGE

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

[3.6.D.1] LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

[Doc M1] a. No pressure boundary LEAKAGE;

[3.6.D.1.a] b. ≤ 5 gpm unidentified LEAKAGE; [and]

[3.6.D.1.c] c. ≤ [30] gpm total LEAKAGE averaged over the previous 24 hour period; [and]

[3.6.D.1.b] d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous [4] hour period in MODE 1.[]

CLB3 25 24 CLB1

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.6.D.2]	A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
[3.6.D.3]	B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce unidentified LEAKAGE to within limits. OR	4 hours (continued)

increase PA1

CTS

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.6.D.3]	B. (continued)	B.2 Verify source of unidentified LEAKAGE increase is not service sensitive type 304 or type 316 austenitic stainless steel.	4 hours
[3.6.D.2] [3.6.D.3] [DOC M1]	C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
[4.6.D.1]	SR 3.4.4.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	~~8~~ 4 hours CLB 2

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.4

RCS Operational LEAKAGE

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

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.4 - RCS OPERATIONAL LEAKAGE

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The limit on the rate of increase in unidentified LEAKAGE is modified to reflect current licensing basis, which is based on conformance to the Staff Position on Leak Detection in Generic Letter 88-01, Supplement 1.

CLB2 The surveillance frequency has been maintained in accordance with the current licensing bases based and the experience at JAFNPP with respect to capabilities of the equipment to detect and alarm increased LEAKAGES.

CLB3 The bracketed total leakage limit has been added based on the current requirements in CTS 3.6.D.1.c.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The wording of Required Action B.1 is modified to maintain consistency with the wording of Condition B. The intent of Required Action B.1 (to reduce "unidentified LEAKAGE increase" as opposed to reducing all LEAKAGE to within limits) is clearly presented in the Bases. Furthermore, once the condition of "unidentified leakage increase not within limit" is corrected, Condition B is exited and Required Action B.1 no longer applies. Any other LEAKAGE limit exceeded will continue to be addressed by Condition A and its Required Actions. As such, this is an editorial clarification consistent with the intent and the ISTS rules of usage.

PA1 3.4.4-01

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

None

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.4

RCS Operational LEAKAGE

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Operational LEAKAGE

BASES

BACKGROUND

The RCS includes systems and components that contain or transport the coolant to or from the reactor core. The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary (RCPB). The joints of the RCPB components are welded or bolted.

Some joints in ≤1" piping are also threaded. DB3

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. Limits on RCS operational LEAKAGE are required to ensure appropriate action is taken before the integrity of the RCPB is impaired. This LCO specifies the types and limits of LEAKAGE. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and GDC/55 of 10 CFR 50, Appendix A (Refs 1, 2, and 3). DB1 UFSAR, Section 16.6 PA2

The safety significance of RCS LEAKAGE from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the primary containment drywell is necessary. Methods for quickly separating the identified LEAKAGE from the unidentified LEAKAGE are necessary to provide the operators quantitative information to permit them to take corrective action should a leak occur that is detrimental to the safety of the facility or the public. PA1

A limited amount of leakage inside primary containment drywell is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the primary containment atmosphere, if possible, so as not to mask RCS operational LEAKAGE detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The allowable RCS operational LEAKAGE limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background LEAKAGE due to equipment design and the detection capability of the instrumentation for determining system LEAKAGE were also considered. The evidence from experiments suggests that, for LEAKAGE even greater than the specified unidentified LEAKAGE limits, the probability is small that the imperfection or crack associated with such LEAKAGE would grow rapidly.

The unidentified LEAKAGE flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs (Refs. ~~2~~ 4 and ~~3~~ 5) shows that leakage rates of hundreds of gallons per minute will precede crack instability (~~Ref. 4~~ Refs. 6 and 7). DB2

The low limit on increase in unidentified LEAKAGE assumes a failure mechanism of intergranular stress corrosion cracking (IGSCC) in service sensitive type 304 and type 316 austenitic stainless steel that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration. PA2

No applicable safety analysis assumes the total LEAKAGE limit. The total LEAKAGE limit considers RCS inventory makeup capability and drywell floor sump capacity.

RCS operational LEAKAGE satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 8). X1

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, ~~being~~ because it is indicative of material degradation. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. PA2

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

The 5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring, drywell sump level monitoring, and containment air cooler condensate flow rate monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB.

drywell floor drain sump monitoring system

DB3

c. Total LEAKAGE

The total LEAKAGE limit is based on a reasonable minimum detectable amount. The limit also accounts for LEAKAGE from known sources (identified LEAKAGE). Violation of this LCO indicates an unexpected amount of LEAKAGE and, therefore, could indicate new or additional degradation in an RCPB component or system.

which may be detected by the drywell equipment drain sump monitoring system

DB3 PA2

d. Unidentified LEAKAGE Increase

An unidentified LEAKAGE increase of > 2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the LEAKAGE. The increase is measured relative to the steady state value; temporary changes in LEAKAGE rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable in MODE 1 when operating pressures and temperatures are established. Violation of this LCO could result in continued degradation of the RCPB.

CLB1

APPLICABILITY

In MODES 1, 2, and 3, the RCS operational LEAKAGE LCO applies, because the potential for RCPB LEAKAGE is greatest when the reactor is pressurized.

In MODES 4 and 5, RCS operational LEAKAGE limits are not required since the reactor is not pressurized and stresses in the RCPB materials and potential for LEAKAGE are reduced.

(continued)

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE; however, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

An unidentified LEAKAGE increase of > 2 gpm within a 4 hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the LEAKAGE rate such that the current rate is less than the "2 gpm increase in the previous 14 hours" limit; either by isolating the source or other possible methods) is to evaluate service sensitive type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased LEAKAGE. This type piping is very susceptible to IGSCC.

The 4 hour Completion Time is reasonable to properly reduce the LEAKAGE increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

The RCS LEAKAGE is monitored by a variety of instruments designed to provide alarms when LEAKAGE is indicated and to quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.6, "RCS Leakage Detection Instrumentation." Sump level and flow rate are typically monitored to determine actual LEAKAGE rates; however, any method may be used to quantify LEAKAGE within the guidelines of Reference 5. In conjunction with alarms and other administrative controls, an 8 hour Frequency for this Surveillance is appropriate for identifying LEAKAGE and for tracking required trends (Ref. 6).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. GEAP-5620, April 1968.
3. NUREG-76/067, October 1975.
4. FSAR, Section [5.2.7.5.2].
5. Regulatory Guide 1.45.
6. Generic Letter 88-01, Supplement 1.

Insert Ref DB2

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. UFSAR, Section 16.6.
4. GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flows, General Electric Company, April 1968.
5. NUREG-75/067, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping in Boiling Water Reactors, October 1975.
6. UFSAR, Section 4.10.
7. UFSAR, Section 16.3.
8. 10 CFR 50.36(c)(2)(ii).
9. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.
10. Generic Letter 88-01, NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping, US Nuclear Regulatory Commission, January 1988.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.4

RCS Operational LEAKAGE

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

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.4 - RCS OPERATIONAL LEAKAGE

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The limit on the rate of increase in unidentified LEAKAGE is modified to reflect current licensing basis, which is based on conformance to the Staff Position on Leak Detection in Generic Letter 88-01, Supplement 1.

CLB2 The surveillance frequency has been maintained in accordance with the current licensing bases based and the experience at JAFNPP with respect to capabilities of the equipment to detect and alarm increased LEAKAGES.

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 changes have been made for clarification, correction, or improvement with no change in intent.

PA3 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage," has not been incorporated in ITS. Subsequent ITS Specifications and Bases renumbered accordingly.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 JAFNPP was designed and under construction prior to the promulgation of Appendix A to 10 CFR 50 - General Design Criteria for Nuclear Power Plants. The JAFNPP Construction Permit was issued on May 20, 1970. The proposed General Design Criteria (GDC) were initially published for comment in the Federal Register on July 11, 1967 (32 FR 10213) and published in final form in the Federal Register on February 20, 1971 (36 FR 3256), and amended on July 7, 1971 (36 FR 12733). UFSAR Section 16.6, "Conformance to AEC Design Criteria," describes the JAFNPP current licensing basis with regard to the GDC. ISTS statements concerning the GDC are modified in the ITS to reference UFSAR Section 16.6.

DB2 References have been revised to reflect plant specific References. References have been renumbered to reflect this change.

DB3 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific design.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.4 - RCS OPERATIONAL LEAKAGE

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN 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: 3.4.4

RCS Operational LEAKAGE

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE;

c. ≤ 25 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce unidentified LEAKAGE increase to within limits. OR	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Verify source of unidentified LEAKAGE increase is not service sensitive type 304 or type 316 austenitic stainless steel.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	4 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Operational LEAKAGE

BASES

BACKGROUND

The RCS includes systems and components that contain or transport the coolant to or from the reactor core. The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary (RCPB). The joints of the RCPB components are welded or bolted. Some joints in ≤ 1" piping are also threaded.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. Limits on RCS operational LEAKAGE are required to ensure appropriate action is taken before the integrity of the RCPB is impaired. This LCO specifies the types and limits of LEAKAGE. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and UFSAR, Section 16.6 (Refs. 1, 2, and 3).

The safety significance of RCS LEAKAGE from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the drywell is necessary. Methods for quickly separating the identified LEAKAGE from the unidentified LEAKAGE are necessary to provide the operators quantitative information to permit them to take corrective action should a leak occur that is detrimental to the safety of the facility or the public.

A limited amount of leakage inside the drywell is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the primary containment atmosphere, if possible, so as not to mask RCS operational LEAKAGE detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The allowable RCS operational LEAKAGE limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background LEAKAGE due to equipment design and the detection capability of the instrumentation for determining system LEAKAGE were also considered. The evidence from experiments suggests that, for LEAKAGE even greater than the specified unidentified LEAKAGE limits, the probability is small that the imperfection or crack associated with such LEAKAGE would grow rapidly.

The unidentified LEAKAGE flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs (Refs. 4 and 5) shows that leakage rates of hundreds of gallons per minute will precede crack instability (Refs. 6 and 7).

The low limit on increase in unidentified LEAKAGE assumes a failure mechanism of intergranular stress corrosion cracking (IGSCC) in service sensitive type 304 and type 316 austenitic stainless steel that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration.

No applicable safety analysis assumes the total LEAKAGE limit. The total LEAKAGE limit considers RCS inventory makeup capability and drywell floor sump capacity.

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, because it is indicative of material degradation. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

The 5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the drywell floor drain sump monitoring system can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB.

c. Total LEAKAGE

The total LEAKAGE limit is based on a reasonable minimum detectable amount. The limit also accounts for LEAKAGE from known sources (identified LEAKAGE which may be detected by the drywell equipment drain sump monitoring system). Violation of this LCO indicates an unexpected amount of LEAKAGE and, therefore, could indicate new or additional degradation in an RCPB component or system.

d. Unidentified LEAKAGE Increase

An unidentified LEAKAGE increase of > 2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the LEAKAGE. The increase is measured relative to the steady state value; temporary changes in LEAKAGE rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable in MODE 1 when operating pressures and temperatures are established. Violation of this LCO could result in continued degradation of the RCPB.

APPLICABILITY

In MODES 1, 2, and 3, the RCS operational LEAKAGE LCO applies, because the potential for RCPB LEAKAGE is greatest when the reactor is pressurized.

In MODES 4 and 5, RCS operational LEAKAGE limits are not required since the reactor is not pressurized and stresses in the RCPB materials and potential for LEAKAGE are reduced.

(continued)

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE; however, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

An unidentified LEAKAGE increase of > 2 gpm within a 24 hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the LEAKAGE rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to evaluate service sensitive type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased LEAKAGE. This type of piping is very susceptible to IGSCC.

The 4 hour Completion Time is reasonable to properly reduce the LEAKAGE increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

The RCS LEAKAGE is monitored by a variety of instruments designed to provide alarms when LEAKAGE is indicated and to quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.5, "RCS Leakage Detection Instrumentation." Sump level and flow rate are typically monitored to determine actual LEAKAGE rates; however, any method may be used to quantify LEAKAGE within the guidelines of Reference 9. In conjunction with alarms and other administrative controls, a 4 hour Frequency for this Surveillance is appropriate for identifying LEAKAGE and for tracking required trends (Ref. 10).

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. UFSAR, Section 16.6.
4. GEAP-5620, Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flows, General Electric Company, April 1968.
5. NUREG-75/067, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping in Boiling Water Reactors, October 1975.
6. UFSAR, Section 4.10.
7. UFSAR, Section 16.3.
8. 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

REFERENCES (continued)

9. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.

10. Generic Letter 88-01, NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping, US Nuclear Regulatory Commission, January 1988.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.5

RCS Leakage Detection Instrumentation

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

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.5

RCS Leakage Detection Instrumentation

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.4.5

A1

ITS

3.2 (cont'd)

RCS

3.4.5 E. Drywell Leak Detection

age

Instrumentation

A2

[LCO 3.4.5] The limiting conditions for operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-5.

add LCO

A2

F. Feedwater Pump Turbine and Main Turbine Trip

The limiting conditions for operation for the instrumentation that provides a feedwater pump turbine and main turbine trip are given in Table 3.2-6.

G. Recirculation Pump Trip

The limiting conditions for operation for the instrumentation that trip(s) the recirculation pumps as a means of limiting the consequences of a failure to scram during an anticipated transient are given in Table 3.2-7.

H. Accident Monitoring Instrumentation

The limiting conditions for operation for the instrumentation that provides accident monitoring are given in Table 3.2-8.

I. 4kv Emergency Bus Undervoltage Trip

The limiting conditions for operation for the instrumentation that prevents damage to electrical equipment or circuits as a result of either a degraded or loss-of-voltage condition on the emergency electrical buses are given in Table 3.2-2.

ITS

4.2 (cont'd)

[3.4.5] E. Drywell Leak Detection

RCS

age

Instrumentation

[SR 3.4.5.2] [SR 3.4.5.3] Instrumentation shall be calibrated and checked as indicated in Table 4.2-5.

A2

F. Feedwater Pump Turbine and Main Turbine Trip

Instrumentation shall be tested and calibrated as indicated in Table 4.2-6.

See ITS: 3.3.2.2

G. Recirculation Pump Trip

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-7.

System logic shall be functionally tested as indicated in Table 4.2-7.

See ITS: 3.3.4.1

H. Accident Monitoring Instrumentation

Instrumentation shall be demonstrated operable by performance of a channel check, channel calibration and functional test as indicated in Table 4.2-8, as applicable.

See ITS: 3.3.3.1

I. Not Used

See ITS: 3.3.8.1

Specification 3.4.5

A1

3.6 (cont'd)

2. With reactor coolant system leakage greater than the limits specified in 3.6.D.1.a or 3.6.D.1.c, the leakage rate shall be reduced to within these limits within 4 hours or the reactor shall be in at least the hot standby condition within the following 12 hours and in cold condition within the next 24 hours.

3. With an increase in unidentified reactor coolant system leakage equal to or greater than the limit specified in 3.6.D.1.b, the source of the leakage shall be identified within 4 hours or the reactor shall be in at least hot standby condition within the next 12 hours and in cold condition within the following 24 hours.

[LCO 3.4.5]

Drywell

4. The Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring and Floor Drain Sump Monitoring) and the Continuous Atmosphere Monitoring System (Gaseous and Particulate) shall be operable when the reactor coolant leakage limits of Specification 3.6.D.1 are in effect.

[Applicability]

one channel of

L1

MODES 1, 2, and 3

4.6 (cont'd)

2. Not Used

See ITS 3.4.4

3. Not Used

L5

Drywell

4. The Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring and Floor Drain Sump Monitoring) instrumentation shall be calibrated and checked as specified in Surveillance Requirement 4.2.E. Continuous Atmosphere Monitoring System (Gaseous and Particulate) instrumentation shall be functionally tested and calibrated as specified in Table 4.6-2.

[SR 3.4.5.1]
[SR 3.4.5.2]
[SR 3.4.5.3]

A1

A2

TABLE 3.2-5

INSTRUMENTATION THAT MONITORS LEAKAGE DETECTION INSIDE THE DRYWELL

Minimum No. of Operable Instrument Channels	Instrument (1)	Action (2)
1	Equipment Drain Sump Flow Integrator	A
	and	
[LCO 3.4.5.a] 1	~~Floor Drain Sump Flow Integrator~~ Drywell floor drain sump monitoring system	

L5

LA1

NOTES FOR TABLE 3.2-5

LA2

1. The two flow integrators, one for the equipment drain sump and the other for the floor drain sump, comprise the Basic Instrument System.

A3

2. Action

   A. Refer to Specification 3.6.D.

75

Amendment No. 36

A1

L5

3.6 (cont'd)

4.6 (cont'd)

add — NOTE — — — LCO 3.0.4 is not applicable

L3

[ACTION A] 5. With the Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring or Floor Drain Sump Monitoring) inoperable, restore the system to operable status within 24 hours or be in at least hot shutdown within the next 12 hours and in the cold condition within the following 24 hours.

Drywell

30 days

L2

[ACTION C]

[ACTION B] 6. With the Continuous Atmosphere Monitoring System (gaseous) or the Continuous Atmosphere Monitoring System (particulate) inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours. Otherwise be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

L1

L3

add
----- Note - - -
LCO 3.0.4 is not applicable

[ACTION C]

Add ACTION D

M2

Specification 3.4.5

A1

JAFNPP

A2

Table 4.6-2

Minimum Test and Calibration Frequency for Continuous Atmosphere Monitoring System

ITS →

	Inst. Channel	Inst. Functional Test [SR 3.4.5.2]	Calibration [SR 3.4.5.3]	Sensor Check [SR 3.4.5.1]
LCO 3.4.5.b	1. Air Particulate Analyzer	~~None~~ 31 days	Once / 3 mos.	once / ~~day~~ 12 hours
	2. Gaseous Activity Analyzer	~~None~~ 31 days	Once / 3 mos.	once / ~~day~~ 12 hours

Channel

A1

L1

M1

M1

Specification 3.4.5 A1

A2

TABLE 4.2-5

MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION

ITS →

Instrument Channel	[SR 3.4.5.2] Instrument Functional Test	[SR 3.4.5.3] Calibration Frequency	~~Instrument Check (Note 4)~~ A4
1) Equipment Drain Sump Flow Integrator	(Note 1)	Q	~~D~~ L5
2) Floor Drain Sump Flow Integrator LA1	(Note 1)	Q	~~D~~ L4

NOTE: See notes following Table 4.2-5.

Specification 3.4.5

A1

ITS

NOTES FOR TABLES 4.2-1 THROUGH 4.2-5

A5

[SR 3.4.5.2]

1. ~~Initially once every month until acceptance failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instruments operate in a environment similar to that of JAFNPP.~~

2. Functional tests are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed within seven (7) days prior to each startup.

3. Calibrations are not required when these instruments are not required to be operable or are tripped. Calibration tests shall be performed within seven (7) days prior to each startup or prior to a pre-planned shutdown.

A4

4. ~~Instrument checks are not required when these instruments are not required to be operable or are tripped.~~

5. This instrumentation is exempt from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

6. These instrument channels will be calibrated using simulated electrical signals once every three months.

7. Simulated automatic actuation shall be performed once per 24 months.

8. Reactor low water level, and high drywell pressure are not included on Table 4.2-1 since they are listed on Table 4.1-2.

9. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

See 3.3

10. (Deleted)

11. Perform a calibration once per 24 months using a radiation source. Perform an instrument channel alignment once every 3 months using a current source.

12. (Deleted)

13. (Deleted)

14. (Deleted)

15. Sensor calibration once per 24 months. Master/slave trip unit calibration once per 6 months.

16. The quarterly calibration of the temperature sensor consists of comparing the active temperature signal with a redundant temperature signal.