VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 18, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	01-281
CM/RAB	R0
Docket Nos.	50-338
	50-339
License Nos.	NPF-4
	NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED IMPROVED TECHNICAL SPECIFICATIONS
REQUEST FOR ADDITIONAL INFORMATION – SECTIONS 3.4 & 3.6

This letter transmits responses to the NRC's request for additional information regarding Sections 3.4 and 3.6 of the North Anna Power Station Units 1 and 2 proposed Improved Technical Specifications (ITS). The North Anna ITS license amendment request was submitted to the NRC in a December 11, 2000 letter (Serial No. 00-606). The NRC requested additional information on ITS Sections 3.4 and 3.6 in a letter dated April 23, 2001 (TAC Nos. MB0799 and MB0800).

The attachment includes each NRC question, the response to each question, and the required revisions to the original ITS license amendment request, based on the response to each question.

If you have any further questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachment

Commitments made in this letter: None

A001

cc: U.S. Nuclear Regulatory Commission
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North Anna Power Station

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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 18TH day of June, 2001.

My Commission Expires: May 31, 2002.

Vicki L. Hull
Notary Public

(SEAL)

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.1 RCS Pressure, Temperature, and Flow DNB Limits

3.4-01 ITS (SR) 3.4.1.4
Standard Technical Specification (STS) SR 3.4.1.4 Note
Current Technical Specifications (CTS) SR 4.2.5.2
Justification for Deviation (JFD) 2

NRC RAI: STS (SR) 3.4.1.4 has a note which states that the surveillance is not required to be performed until 24 hours after greater than or equal to 90 percent Rated Thermal Power (RTP). This Note allows entrance into the APPLICABILITY statement of the Limiting Condition for Operation (LCO) (i.e. Mode 1) without the performance of the surveillance. JFD 2 states that it is not necessary to specify a Frequency beyond 18 months. **Comment:** STS SR 3.4.1.4 note should be incorporated into ITS SR 3.4.1.4.

Response: The Company will take the action proposed in the Comment, with certain modifications. STS 3.4.1.4 requires verification by precision heat balance that RCS total flow rate is ≥ 295,000 gpm and greater than or equal to the limit specified in the COLR every 18 months. The STS SR is modified by a Note that states, "Not required to be performed until 24 hours after ≥ 90% RTP." This Note performs two purposes. First, it allows entry into MODE 1 if the Surveillance has not been met. Without the Note, if the Surveillance has not been met within its Frequency (18 months + 25%, or 22.5 months), entry into MODE 1 without first performing the Surveillance is prohibited by SR 3.0.4. As stated in JFD 2, this test is currently performed after each startup, typically a few weeks after achieving full power. The second purpose of the Note is to specify a maximum time after reaching full power following refueling before performing the Surveillance. The Note allows 24 hours after exceeding 90% RTP. As described in JFD 2, this restriction cannot be met at North Anna. Establishment of the conditions for performance of the precision heat balance is time consuming and requires installation of equipment and establishment of stable operating conditions. Twenty-four hours after exceeding 90% RTP does not allow sufficient time to establish stable plant conditions, install the instrumentation, perform the test, and analyze the results. Therefore, the Note has been modified to state, "Not required to be performed until 30 days after ≥ 90% RTP." This provides time to reach a stable operating condition after startup, install the necessary equipment, perform the test, and analyze the results. See also the response to Question 3.4-15.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is ≥ 295,000 gpm and is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.4	------------------NOTE------------------ Not required to be performed until 30 days after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 295,000 gpm and is greater than or equal to the limit specified in the COLR.	 18 months

RAI 3.4-01 R1

CTS

SURVEILLANCE REQUIREMENTS

CTS	SURVEILLANCE		FREQUENCY	
4.2.5.1	SR 3.4.1.1	Verify pressurizer pressure is ~~≥ [2200] psig~~.	12 hours	TSTF-339
4.2.5.1	SR 3.4.1.2	Verify RCS average temperature is ~~≤ [581]°F~~. is less than or equal to the limit specified in the COLR	12 hours	
4.2.5.1	SR 3.4.1.3	Verify RCS total flow rate is ≥ ~~[284,000]~~ 295,000 gpm and	12 hours	1
new 4.2.5.2	SR 3.4.1.4	-----NOTE----- Not required to be performed until ~~24 hours~~ 30 days after ≥ ~~[90]~~% RTP. ------------ Verify by precision heat balance that RCS total flow rate is ≥ ~~[284,000]~~ 295,000 gpm and	2 1 ~~[~~18~~]~~ months	RAI 3.4-01 R1 1 1 TSTF-339

greater than or equal to the limit specified in the COLR

Rev. 1

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ITS SR 3.4.1.4 requires verification of RCS total flow rate and has a Frequency of 18 months. It contains a Note which states, "Not required to be performed until 24 hours after ≥ 90% RTP." CTS Surveillance Requirement 4.2.5.2 also requires a verification of RCS total flow rate every 18 months but does not contain the equivalent of the ITS SR Note. The ITS SR 3.4.1.4 Note is modified to state, "Not required to be performed until 30 days after ≥ 90% RTP." The Note is required in order to enter MODE 1 in the unlikely event the Surveillance Frequency expires while not in MODE 1. Establishment of the conditions for performance of the test is time consuming and the test is typically not performed until several weeks after startup following refueling. The 30 day period after exceeding 90% RTP is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. This is acceptable because the calibration values used for the RCS flow indications until the test can be performed are verified by trending plant parameters, such as electrical output, monitoring of individual pieces of the flow calorimetric, and cognizance of design changes which could affect RCS flow.

RAI 3.4-01 R1

ITS 3.4.1

A.1

11-26-77

ITS

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

LCO 3.4.1

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

Appl

APPLICABILITY: MODE 1

ACTION:

Action A
Action B

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1
SR 3.4.1.2
SR 3.4.1.3

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

SR 3.4.1.4

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

SR 3.4.1.4 Note

Insert proposed ITS SR 3.4.1.4 Note

L.1

RAI 3.4-01 R1

Rev. 1

A.1

ITS 3.4.1

ITS

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

LCO 3.4.1

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

Appl

APPLICABILITY: MODE 1

ACTION:

Action A

Action B

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SR 3.4.1.1
SR 3.4.1.2
SR 3.4.1.3

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

SR 3.4.1.4

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

SR 3.4.1.4 Note

Insert proposed ITS SR 3.4.1.4 Note

L.1

RAI 3.4-01 R1

Rev. 0

The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits From the Technical Specifications, that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. NRC-approved Topical Report WCAP-14483, "Generic Methodology for Expanded Core Operating Limits Report" determined that the specific values for the DNB parameters may be relocated to the COLR as long as the limiting RCS total flow limit is retained in the LCO. The LCO continues to require that the core be operated within the DNB limits. The methodologies used to develop the DNB parameters in the COLR have obtained prior approval by the NRC in accordance with Generic Letter 88-16. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, Core Operating Limits Report. ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.2.5.2 states that the Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months. ITS SR 3.4.1.4 requires measurement of the RCS total flow rate every 18 months and is modified by a Note which states, "Not required to be performed until 30 days after ≥ 90% RTP." This changes the CTS by relaxing the Surveillance Frequency in order to allow entry into MODE 1 to perform the test and requires the test to be performed within 30 days after exceeding 90% RTP.

RAI 3.4-01 R1

The purpose of CTS 4.2.5.2 is to accurately determine the RCS total flow rate. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. An accurate measurement of the RCS total flow rate must be performed at full power under stable operating conditions. In the unlikely event that the Surveillance Frequency has expired while the unit is not in MODE 1, SR 3.0.4 would prevent entry into MODE 1 to perform the test. Therefore, without the Note the Surveillance would have to be performed prior to entering MODE 1 and, in all likelihood, performed again in MODE 1 to obtain accurate results. Therefore, the Note allowance to enter MODE 1 in order to perform the test may result in performing the test less frequently. The Note also applies a 30 day period after exceeding 90% RTP to perform the test. This is a reasonable period to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. This analyze the results. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS. RAI 3.4-01 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.2 RCS Minimum Temperature for Criticality

3.4-02 ITS 3.4.2
JFD 2

NRC RAI: JFD 2 for ITS 3.4.2 states that editorial changes were made for enhanced clarity or to be consistent with the ISTS Writers Guide. The markup copy of ITS 3.4.2 does not have "2" listed. **Comment:** Provide revised markup with appropriate placement of "2".

Response: The Company will take the action proposed in the Comment, with certain modifications. JFD 2 does not apply to ITS 3.4.2 and will be deleted from the list of JFDs for ITS 3.4.2.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.2, RCS MINIMUM TEMPERATURE FOR CRITICALITY

1. The brackets are removed and the proper plant specific information/value is provided.

2. Not used. RAI 3.4-02 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.5 RCS Loops - MODE 3

3.4-03 ITS SR 3.4.5.3
STS SR 3.4.5.3
CTS SR 4.4.1.2.1
JFD 3
Technical Specifications Task Force (TSTF)-265 Rev. 2

NRC RAI: STS SR 3.4.5.3 requires the licensee to verify correct breaker alignment and indicated power are available to each required pump. This wording was approved via TSTF-265 Rev. 2. ITS SR 3.4.5.3 did not adopt TSTF-265 Rev. 2 in its entirety. ITS SR 3.4.5.3 would verify correct breaker alignment and indicated power to the required pump not in operation. The TSTF revised the SR to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating. **Comment:** TSTF-265 Rev. 2 should be adopted in its entirety.

Response: The Company does not agree with the action recommended in the Comment. TSTF-265 revises SR 3.4.5.3 from, "Verify correct breaker alignment and indicated power are available to the required pump that is not in operation," to "Verify correct breaker alignment and indicated power are available to each required pump." The TSTF-265 modifications to the SR would require performance of SR 3.4.5.3 on operating loops when, as stated in the TSTF-265 changes to the Bases, operation is evidence of proper breaker alignment and power availability. As a result, the proposed changes in TSTF-265 add additional administrative burden with no compensatory increase in safety. Therefore, the Company will retain the CTS requirements and will propose a generic change to the ISTS. Attachment 3 of the submittal has been revised to indicate that TSTF-265 was only partially incorporated. The JFD for this change has been expanded to provide additional justification. See also the response to Questions 4, 5, 6, 18, 21, 24, and 26.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.5, RCS LOOPS - MODE 3

1. The brackets are removed and the proper plant specific information/value is provided.

2. NUREG-1431 Specification 3.4.5 contains requirements and actions on the Rod Control System based on the assumption that the accident analysis for an uncontrolled RCCA bank withdrawal requires two RCS loops to be in operation. The North Anna accident analysis for uncontrolled RCCA bank withdrawal from a subcritical condition assumes that only one RCS loop is in operation. As a result, the ITS LCO does not contain requirements on the reactor trip breakers or the Rod Control System. ITS Condition C.1 (ISTS Condition D.1), which requires the CRDMs to be de-energized when no RCS loop is in operation, was retained to protect this analysis assumption. These changes are consistent with the North Anna accident analysis assumptions.

3. TSTF-265 is modified. TSTF-265 expanded the Surveillance to require performance on both the operating and non-operating pump. This portion of the generic change is not adopted and the CTS Surveillance wording is retained. The TSTF-265 change to require verification of breaker position and indicated power availability on the operating pump is not necessary as pump operation is, as stated in the TSTF, an adequate indication of available power. The CTS Surveillance wording adequately verifies compliance with the LCO without the unnecessary administrative burden imposed by the TSTF-265 Surveillance revision. Therefore, the CTS Surveillance wording is retained. RAI 3.4-03 R1

3.4.6 RCS Loops - MODE 4

3.4-04 ITS SR 3.4.6.3
STS SR 3.4.6.3
CTS SR 4.4.1.3.2
JFD 2
TSTF-265 Rev. 2

NRC RAI: STS SR 3.4.6.3 requires the licensee to verify correct breaker alignment and indicated power are available to each required pump. This wording was approved via TSTF-265 Rev. 2. ITS SR 3.4.6.3 did not adopt TSTF-265 Rev. 2 in its entirety. ITS SR 3.4.6.3 would verify correct breaker alignment and indicated power to the required pump not in operation. The TSTF revised the SR to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating. **Comment:** TSTF-265 Rev. 2 should be adopted in its entirety.

Response: The Company does not agree with the action recommended in the Comment. TSTF-265 revises SR 3.4.6.3 from, "Verify correct breaker alignment and indicated power are available to the required pump that is not in operation," to "Verify correct breaker alignment and indicated power are available to each required pump." The TSTF-265 modifications to the SR would require performance of SR 3.4.6.3 on operating loops when, as stated in the TSTF-265 changes to the Bases, operation is evidence of proper breaker alignment and power availability. As a result, the proposed changes in TSTF-265 add additional administrative burden with no compensatory increase in safety. Therefore, the Company will retain the CTS requirements and will propose a generic change to the ISTS. Attachment 3 of the submittal has been revised to indicate that TSTF-265 was only partially incorporated. The JFD for this change has been expanded to provide additional justification. See also the response to Questions 3, 5, 6, 18, 21, 24, and 26.

1. The brackets are removed and the proper plant specific information/value is provided.

2. TSTF-265 is modified. TSTF-265 expanded the Surveillance to require performance on both the operating and non-operating pump. This portion of the generic change is not adopted and the CTS Surveillance wording is retained. The TSTF-265 change to require verification of breaker position and indicated power availability on the operating pump is not necessary as pump operation is, as stated in the TSTF, an adequate indication of available power. The CTS Surveillance wording adequately verifies compliance with the LCO without the unnecessary administrative burden imposed by the TSTF-265 Surveillance revision. Therefore, the CTS Surveillance wording is retained.

RAI 3.4-04 R1

3.4.7 RCS Loops - MODE 5, Loops Filled

3.4-05 ITS SR 3.4.7.3
STS SR 3.4.7.3
CTS SR 4.4.1.3.2
JFD 4
TSTF-265 Rev. 2

NRC RAI: STS SR 3.4.7.3 requires the licensee to verify correct breaker alignment and indicated power are available to each required pump. This wording was approved via TSTF-265 Rev. 2. ITS SR 3.4.7.3 did not adopt TSTF-265 Rev. 2 in its entirety. ITS SR 3.4.7.3 would verify correct breaker alignment and indicated power to the required pump not in operation. The TSTF revised the SR to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating. **Comment:** TSTF-265 Rev. 2 should be adopted in its entirety.

Response: The Company does not agree with the action recommended in the Comment. TSTF-265 revises SR 3.4.7.3 from, "Verify correct breaker alignment and indicated power are available to the required pump that is not in operation," to "Verify correct breaker alignment and indicated power are available to each required pump." The TSTF-265 modifications to the SR would require performance of SR 3.4.7.3 on operating loops when, as stated in the TSTF-265 changes to the Bases, operation is evidence of proper breaker alignment and power availability. As a result, the proposed changes in TSTF-265 add additional administrative burden with no compensatory increase in safety. Therefore, the Company will retain the CTS requirements and will propose a generic change to the ISTS. Attachment 3 of the submittal has been revised to indicate that TSTF-265 was only partially incorporated. The JFD for this change has been expanded to provide additional justification. See also the response to Questions 3, 4, 6, 18, 21, 24, and 26.

1. The brackets are removed and the proper plant specific information/value is provided.

2. Editorial change made for consistency with other changes made to the ISTS.

3. Editorial change made for enhanced clarify or consistency with the ISTS Writer's Guide.

4. TSTF-265 is modified. TSTF-265 expanded the Surveillance to require performance on both the operating and non-operating pump. This portion of the generic change is not adopted and the CTS Surveillance wording is retained. The TSTF-265 change to require verification of breaker position and indicated power availability on the operating pump is not necessary as pump operation is, as stated in the TSTF, an adequate indication of available power. The CTS Surveillance wording adequately verifies compliance with the LCO without the unnecessary administrative burden imposed by the TSTF-265 Surveillance revision. Therefore, the CTS Surveillance wording is retained. RAI 3.4-05 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

ITS 3.4.8 RCS Loops - MODE 5, Loops Not Filled

3.4-06 ITS SR 3.4.8.2
STS SR 3.4.8.2
CTS SR 4.4.1.3.2
JFD 2
TSTF-265 Rev. 2

NRC RAI: STS SR 3.4.8.2 requires the licensee to verify correct breaker alignment and indicated power are available to each required pump. This wording was approved via TSTF-265 Rev. 2. ITS SR 3.4.8.2 did not adopt TSTF-265 Rev. 2 in its entirety. ITS SR 3.4.8.2 would verify correct breaker alignment and indicated power to the required pump not in operation. The TSTF revised the SR to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating. **Comment:** TSTF-265 Rev. 2 should be adopted in its entirety.

Response: The Company does not agree with the action recommended in the Comment. TSTF-265 revises SR 3.4.8.2 from, "Verify correct breaker alignment and indicated power are available to the required pump that is not in operation," to "Verify correct breaker alignment and indicated power are available to each required pump." The TSTF-265 modifications to the SR would require performance of SR 3.4.8.2 on operating loops when, as stated in the TSTF-265 changes to the Bases, operation is evidence of proper breaker alignment and power availability. As a result, the proposed changes in TSTF-265 add additional administrative burden with no compensatory increase in safety. Therefore, the Company will retain the CTS requirements and will propose a generic change to the ISTS. Attachment 3 of the submittal has been revised to indicate that TSTF-265 was only partially incorporated. The JFD for this change has been expanded to provide additional justification. See also the response to Questions 3, 4, 5, 18, 21, 24, and 26.

1. The brackets are removed and the proper plant specific information/value is provided.

2. TSTF-265 is modified. TSTF-265 expanded the Surveillance to require performance on both the operating and non-operating pump. This portion of the generic change is not adopted and the CTS Surveillance wording is retained. The TSTF-265 change to require verification of breaker position and indicated power availability on the operating pump is not necessary as pump operation is, as stated in the TSTF, an adequate indication of available power. The CTS Surveillance wording adequately verifies compliance with the LCO without the unnecessary administrative burden imposed by the TSTF-265 Surveillance revision. Therefore, the CTS Surveillance wording is retained. RAI 3.4-06 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.11 Pressurizer Power Operated Relief Valves

3.4-07 ITS 3.4.11 ACTIONS F1 & F2
STS 3.4.11 ACTIONS E1 & E2
CTS 3.4.3.2 ACTION A5
JFD 4

NRC RAI: STS 3.4.11 ACTIONS E1 and E2 require that the associated block valve be closed and power removed from the associated block valve if two PORVs are inoperable and not capable of being manually cycled. ITS 3.4.11 did not incorporate these action items. JFD stated that ACTIONS E1 and E2 are not incorporated since they are duplicate actions to ACTIONS C1 and C2. However, CTS 3.4.3.2 ACTION A5 has the same STS 3.4.11 ACTIONS E1 and E2 requirements. **Comment:** STS 3.4.11 ACTIONS E1 and E2 should be incorporated into ITS 3.4.11.

Response: The Company does not agree with the action recommended in the Comment. ITS 3.4.11, Condition C, states that with one PORV inoperable and not capable of being manually cycled, close the associated block valve, remove power from the associated block valve, and restore the PORV to OPERABLE status. The ACTIONS are modified by a Note stating that, "Separate Condition entry is allowed for each PORV and each block valve." ITS 3.4.11, Condition F, applies with "Two PORVs inoperable and not capable of being manually cycled." Under the ITS rules of multiple condition entry (see Example 1.3-5 in ITS 1.3), when Condition F is entered for two PORVs inoperable, Condition C is also entered concurrently for each inoperable valve. Therefore, the Condition C Required Actions C.1 and C.2 are duplicative of the STS ACTIONS E.1 and E.2 (what would be ITS Required Actions F.1 and F.2). As these duplicative Required Actions are unnecessary and confusing, they are removed. In the CTS these actions are not duplicative, as the CTS does not allow multiple condition entry. In the CTS, Action A.4 for one inoperable PORV is exited and Action A.5 is entered when two PORVs become inoperable. Therefore, these Actions are required in both CTS Actions A.4 and A.5. JFD 4 is expanded to incorporate this additional justification. See also the response to Question 3.4-28.

1. This bracketed requirement is deleted because it is not applicable to North Anna. The following requirements are renumbered, where applicable, to reflect this deletion.

2. The brackets are removed and the proper plant specific information/value is provided.

3. The North Anna PORVs are supplied from both the Instrument Air system and backup nitrogen accumulators. The backup nitrogen accumulators are needed for PORV OPERABILITY. A Condition is added to the ITS for one or more PORVs inoperable due to inoperable backup nitrogen supply and the PORVs capable of being manually cycled. A Surveillance is added to verify the OPERABILITY of the backup nitrogen supply. Subsequent items are renumbered as needed. The wording of SR 3.4.11.3 has been revised to reflect this design.

4. ISTS 3.4.11, Condition B, states that with one PORV inoperable and not capable of being manually cycled, close the associated block valve, remove power from the associated block valve, and restore the PORV to OPERABLE status. The ACTIONS are modified by a Note stating that, "Separate Condition entry is allowed for each PORV and each block valve." ISTS 3.4.11, Condition E, applies with "Two PORVs inoperable and not capable of being manually cycled." Under the ITS rules of multiple condition entry (see Example 1.3-5 in ITS 1.3), when Condition E is entered for two PORVs inoperable, Condition B is also entered concurrently for each inoperable valve. Therefore, ISTS Condition B Required Actions B.1 and B.2 are duplicative of the ISTS ACTIONS E.1 and E.2. As these duplicative Required Actions are unnecessary and confusing, they are removed. The North Anna CTS repeats the actions in the condition of two PORVs inoperable and not capable of being manually cycled. However, in the CTS these actions are not duplicative, as the CTS does not allow multiple condition entry. In the CTS, Action A.4 for one inoperable PORV is exited and Action A.5 is entered when two PORVs become inoperable. RAI 3.4-07 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.11 Pressurizer Power Operated Relief Valves

3.4-08 ITS SR 3.4.11.4
STS SR 3.4.11.3
CTS SR 4.4.3.2.1.b.2

NRC RAI: STS SR 3.4.11.3 specifies the performance of a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems. This wording is consistent with CTS SR 4.4.3.2.1.b.2. ITS SR 3.4.11.4 removes the word "air" from the SR. **Comment:** Provide justification for removal of word "air" in SR or make the ITS consistent with the STS and CTS. These changes also affect the ITS SR 3.4.11.4 Bases.

Response: The Company will take the action proposed in the Comment, with certain modifications. As described in JFD 3, the North Anna PORVs are supplied from both the Instrument Air system and backup nitrogen accumulators. Therefore, there are both solenoid air control valves and solenoid nitrogen control valves. The intent of SR 3.4.11.3 is to verify the OPERABILITY of the control valves and check valves used to actuate the PORVs. Requiring only the testing of the "air" control valves in SR 3.4.11.3 would exclude the testing of the nitrogen control valves. This is inappropriate. In order to specify the testing of both the air and nitrogen control valves, the word "air" was deleted and the SR requires testing of "each solenoid control valve." The specification of "solenoid air control valves" in the CTS is an oversight which is corrected in the ITS. An "M" DOC will be added to the CTS to justify the deletion of the word "air" from the Surveillance.

(A.1)

ITS 3.4.11

03-02-99

REACTOR COOLANT SYSTEM

SAFETY AND RELIEF VALVES – OPERATING

ITS

RELIEF VALVES

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

B. Block Valves: Insert proposed Action D Note (L.2)

Action D, E

1. With one block valve inoperable, within 1 hour either restore the block valve to OPERABLE status or place its associated PORV in manual control; restore the block valve to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Insert proposed Action G Note (L.2)

Action G,

2. With both block valves inoperable, ~~within 1 hour either restore the block valves to OPERABLE status or place the PORVs in manual control;~~ restore at least one block valve to OPERABLE status within ~~the next~~ 2 hour; ~~restore the remaining inoperable block valve to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.~~ (A.4) (A.4)

Action note 2

3. The provisions of Specification 3.0.4 are not applicable. ← Move to Note 2 for Actions

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 ~~In addition to the requirements of Specification 4.0.5,~~ each PORV shall be demonstrated OPERABLE: (A.1)

a. At least once per 31 days by performing a CHANNEL FUNCTIONAL TEST, excluding valve operation, and ← See ITS 3.3.1

b. At least once per 18 months by:

SR 3.4.11.3

1. Operating the PORV through one complete cycle of full travel ~~during MODES 3 or 4, and~~ (LA.1)

SR 3.4.11.4

2. Operating the solenoid ~~air~~ control valves and check valves on the associated accumulators in the PORV control systems through one complete cycle of full travel, and (M.1) RAI 3.4-08 R1

3. Performing a CHANNEL CALIBRATION of the actuation instrumentation. See ITS 3.3.1

SR 3.4.11.1

c. At least once per 7 days by verifying that the pressure in the PORV nitrogen accumulators is greater than the surveillance limit.

SR 3.4.11.2

4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION A.4 or A.5 in Specification 3.4.3.2. ← This Surveillance is only required to be met in MODES 1 and 2 (L.1)

A.1

ITS 3.4.11

03-02-99

REACTOR COOLANT SYSTEM

SAFETY AND RELIEF VALVES – OPERATING

RELIEF VALVES

ITS

LIMITING CONDITION FOR OPERATION

ACTION:(Continued) Insert proposed Action D Note L.2

B. Block Valves:

Action D, E

1. With one block valve inoperable, within 1 hour either restore the block valve to OPERABLE status or place its associated PORV in manual control; restore the block valve to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. L.2

Insert proposed Action G Note

Action G, H

2. With both block valves inoperable, ~~within 1 hour either restore the block valves to OPERABLE status or place the PORVs in manual control;~~ restore at least one block valve to OPERABLE status within ~~the next hour~~ 2 hours; ~~restore the remaining inoperable block valve to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.~~ A.4 A.4

Action Note 2

3. The provisions of Specification 3.0.4 are not applicable. Move to Note 2 for Actions

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 ~~In addition to the requirements of Specification 4.0.5,~~ each PORV shall be demonstrated OPERABLE: A.1

a. At least once per 31 days by performing a CHANNEL FUNCTIONAL TEST, excluding valve operation, and See ITS 3.3.1

SR 3.4.11.3

b. At least once per 18 months by:

1. Operating the PORV through one complete cycle of full travel ~~during MODES 3 or 4, and~~ LA.1

SR 3.4.11.4

2. Operating the solenoid ~~air~~ control valves and check valves on the associated accumulators in the PORV control systems through one complete cycle of full travel, and M.1 RAI 3.4-08 R1

3. Performing a CHANNEL CALIBRATION of the actuation instrumentation. See ITS 3.3.1

SR 3.4.11.1

c. At least once per 7 days be verifying that the pressure in the PORV nitrogen accumulators is greater than the surveillance limit.

SR 3.4.11.2

4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION A.4 or A.5 in Specification 3.4.3.2.

This Surveillance is only required to be met in MODES 1 and 2. L.1

A.4 CTS 3.4.3.2, Action B.1, states that with one block valve inoperable, within 1 hour either restore the block valves to OPERABLE status or place the PORVs in manual control; restore the block valve to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. CTS 3.4.3.2, Action B.2, states that with both block valves inoperable, within 1 hour either restore the block valves to OPERABLE status or place the PORVs in manual control; restore at least one block valve to OPERABLE status within the next hour, and restore the remaining inoperable block valve to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. ITS 3.4.11, Action D, states that with one block valve inoperable, place the associated PORV in manual control and restore the block valve to OPERABLE status within 72 hours. ITS 3.4.11, Action G, states that with two block valves inoperable, restore one block valve to OPERABLE status within 2 hours. This changes the CTS by eliminating the actions for one block valve inoperable in the Condition for two block valves inoperable.

This change is acceptable because the requirements have not changed. Under the rules of the ITS, all applicable Conditions are entered. Therefore, with two block valves inoperable, the Conditions and Required Actions for one block valve inoperable must also be followed. As a result, it is not necessary to repeat those Required Actions in the Condition for two block valves inoperable. This change is designated as administrative as it is a change required by the ITS usage rules that does not result in a technical change to the specifications.

MORE RESTRICTIVE CHANGES

M.1 CTS 4.4.3.2.1.b.2 requires operating the solenoid air control valves and check valves on the associated accumulators in the PORV control systems through one complete cycle of full travel every 18 months. ITS SR 3.4.11.4 requires performing a complete cycle of each solenoid control valve and check valve for the accumulators in the PORV control systems every 18 months. This changes the CTS by specifying that each solenoid control valve and check valve in the normal air and backup nitrogen PORV control systems must be tested every 18 months.

RAI 3.4-08 R1

The purpose of the CTS Surveillance is to verify that the PORVs are OPERABLE. The North Anna PORVs are supplied from both the Instrument Air system and backup nitrogen accumulators. The backup nitrogen accumulators are needed for PORV OPERABILITY. This change is acceptable because the ITS Surveillance will verify the OPERABILITY of both the normal air and backup nitrogen PORV supplies in order to verify PORV OPERABILITY. This change is designated as more restrictive because it expands the applicability of a Surveillance.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.12 Low Temperature Overpressure Protection (LTOP) System

3.4-09 ITS 3.4.12 Note 2
STS 3.4.12 Note 2
CTS LCO 3.4.9.3
JFD 6
TSTF-285 Rev. 1

NRC RAI: ITS 3.4.12 Note 2 does not incorporate the changes made by TSTF-285 Rev. 1. Specifically, TSTF-285 Rev. 1 revised the note to state that the accumulator may be unisolated when accumulator pressure is less than the maximum reactor coolant system (RCS) pressure. JFD 6 did not address why TSTF-285 Rev. 1 was not incorporated. Attachment 3 of the North Anna submittal indicates that TSTF-285 Rev. 1 was incorporated. **Comment:** Provide justification for not incorporating TSTF-285 Rev. 1 in its entirety or incorporate TSTF-285 Rev. 1 into ITS 3.4.12 Note 2.

Response: The Company will take the action proposed in the comment, with certain modifications. ITS 3.4.12 incorporates the TSTF-285 changes to Note 2, as modified by JFD 6. However, the STS markup does not show TSTF-285 as modifying Note 2. The markup will be modified to show TSTF-285 as changing Note 2. As stated in JFD 6, the North Anna LTOP analysis does not address the injection of an accumulator when the accumulator pressure is below the temperature-dependent maximum allowable RCS pressure but above the PORV lift setting. Therefore, the allowance to unisolate the accumulator is based on the PORV lift pressure. JFD 6 will be modified to specifically address the change to Note 2 and how the wording in TSTF-285, Rev. 1 was modified to be consistent with the North Anna design. Attachment 3 of the submittal, which lists the Travelers which were considered when developing the North Anna ITS, will be modified to show TSTF-285, Rev. 1 as partially incorporated.

CTS

RAI 3.4-12 R1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.5.3, Note #

LCO 3.4.9.3

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of ~~[one]~~ ~~[high pressure injection (HPI)] pump [and]~~ one charging pump(1) and one low head safety injection (LHSI) pump capable of injecting into the RCS and the accumulators isolated, with power removed from the isolation valve operators, and ~~either a or b below.~~ one of the following pressure relief capabilities: (1) TSTF-280 (1)

~~a. Two RCS relief valves, as follows:~~

(a.) ~~(1)~~ Two power operated relief valves (PORVs) with lift settings ~~within the limits specified in the PTLR, or~~ (2) (7)

~~[2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ [436.5] psig and ≤ [463.5] psig, or]~~ (1)

~~[3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ [436.5] psig and ≤ [463.5] psig].~~ TSTF-285

1. Two charging pumps may be made capable of injecting for ≤ 1 hour for pump swap operations.

b. The RCS depressurized and an RCS vent of ≥ ~~[~~2.07~~]~~ square inches. (3) TSTF-243

Applicability 3.4.9.3

APPLICABILITY: MODE 4 when ~~all~~ any RCS cold leg temperature is ≤ ~~[275]~~ 235°F (Unit 1), 270°F (Unit 2) (3)
MODE 5,
MODE 6 when the reactor vessel head is on. (6)

---------------------------------NOTE(5)---------------------------------
(2) Accumulator isolation with power removed from the isolation valve operators is only required when accumulator pressure is greater than or equal to ~~the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.~~ the PORV lift setting TSTF-285 (6)

(7)

of:

1. ≤ 500 psig (Unit 1), 415 psig (Unit 2) when any RCS cold leg temperature ≤ 235°F (Unit 1), 270°F (Unit 2)

2. ≤ 395 psig (Unit 1), 375 psig (Unit 2) when any RCS cold leg temperature ≤ 150°F (Unit 1), 130°F (Unit 2)

or

RAI 3.4-9 R1

Rev. 1

The ISTS requires the accumulators to be isolated when accumulator pressure is greater than the maximum RCS pressure for the existing cold leg temperature as allowed by the P/T limit curves. The ISTS is revised to require the accumulators to be isolated when accumulator pressure is greater than the PORV lift setpoint pressure given in the LCO. The North Anna LTOP analysis does not address the situation of an accumulator injecting with the accumulator pressure above the PORV lift setting but below the maximum RCS pressure for the existing cold leg temperature as allowed by the P/T limit curves. The analysis does not address a PORV being used to relieve pressure from accumulator injection. If the accumulator pressure is below the PORV lift setpoint (which is also below the limiting pressure for the existing cold leg temperature), injection of an accumulator cannot exceed the maximum RCS pressure for the existing conditions. This revised allowance is stated in an LCO Note, a Note to Condition C, and a Note to SR 3.4.12.3.

TSTF-285 modified the ISTS Applicability Note to state that the accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing cold leg temperature as allowed by the P/T limit curves provided in the PTLR, and moved the Applicability Note to an LCO Note. The movement of the Note to an LCO Note has been adopted in the North Anna ITS. However, the wording changes made to the Note in TSTF-285 are not consistent with the North Anna LTOP analysis, as described in the previous paragraph. Therefore, the Note has been revised to be consistent with the North Anna LTOP analysis. RAI 3.4-09 R1

These more stringent controls on accumulator pressure and accumulator isolation will ensure that the assumptions of the North Anna LTOP design are met.

7. Portions of TSTF-280, Revision 1, are not adopted. The revisions to LCO 3.4.12 made by TSTF-280, Revision 1, to clarify the application of the available options are not needed due to the changes made to the LCO to reflect the North Anna analysis and design.

8. The North Anna PORVs are supplied from both the Instrument Air System and backup nitrogen accumulators. The backup nitrogen accumulators are needed for PORV OPERABILITY. A Surveillance is added to verify the OPERABILITY of the backup nitrogen supply. Subsequent items are renumbered as needed.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.12 Low Temperature Overpressure Protection (LTOP) System

3.4-10 ITS 3.4.12 ACTION C
STS 3.4.12 ACTION C
CTS 3.4.9.3
JFD 6
Beyond Scope Item

NRC RAI: JFD 6 states that the North Anna low-temperature overpressure protection (LTOP) design assumes that an accumulator does not inject into the RCS while in the LTOP regime. ITS 3.4.12 ACTION C has been modified to reflect the North Anna LTOP design. However, CTS 3.4.9.3 does not provide restrictions on the accumulator during LTOP applicability. Updated Final Safety analysis Report (UFSAR) Section 5.2.2.2 does not discuss this assumption.
Comment: Provide more information on the North Anna LTOP design assumptions and analyses.

Response: CTS 3.4.9.3 does not provide restrictions on the accumulators during the LTOP applicability because the model specification in NUREG-0452, Revision 3 (Westinghouse Standard Technical Specifications), which was the basis for the current North Anna specifications, did not include controls on accumulators in the LTOP specification. STS 3.4.12, LTOP System, applies restrictions on the accumulators. Therefore, in converting the North Anna CTS to the North Anna ITS, it was necessary to determine the appropriate controls on the accumulators in the LTOP regime. As stated in UFSAR Section 5.2.2.2, "Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analyses." There is an implicit assumption in the LTOP analyses that the accumulators are isolated when accumulator pressure is greater than the PORV lift setting in that there is no evaluation performed to demonstrate that a PORV can maintain RCS pressure below the LTOP limits should an accumulator inject into the RCS at a pressure greater than the PORV lift setting. Therefore, it is necessary to ensure that an accumulator does not inject in these circumstances. This assurance is provided by requiring the accumulators to be isolated and power be removed from the isolation valve operators. This ensures that a single event cannot result in the injection of an accumulator. JFD 6 was expanded to include additional justification.

1. The North Anna LTOP system does not assume the operation of the RHR suction relief valves. References to the RHR suction relief valves are eliminated and the general term "RCS relief valves" is replaced with the more accurate "PORVs" throughout. The North Anna LTOP analysis assumes that only one charging pump and one Low Head Safety Injection (LHSI) pump are available for injection below the LTOP arming temperature. Appropriate changes are made to the ISTS and subsequent items are renumbered or relabeled as necessary.

2. North Anna Power Station is not adopting a Pressure Temperature Limits Report (PTLR) and is retaining the LTOP in the Technical Specifications. References to the PTLR have been deleted.

3. The brackets are removed and the proper plant specific information/value is provided.

4. Changes are made to reflect those changes made to the ISTS.

5. A requirement is added for verification that the PORV keyswitch is in the AUTO position in order to ensure that the LTOP System is activated. This Surveillance and Frequency are consistent with the CTS.

6. The North Anna LTOP design assumes that an accumulator does not inject into the RCS while in the LTOP protection regime. The North Anna CTS does not provide restrictions on the accumulator during LTOP applicability as the existing standard when the North Anna specifications were developed, NUREG-0452, did not address the accumulators in the LTOP range. Therefore, the accumulators were isolated under administrative controls. NUREG-1431 addresses the accumulators in the ISTS 3.4.12. Therefore, appropriate accumulator controls have been added to the North Anna ITS to reflect the North Anna design. RAIs 3.4-10 3.4-11 R1

   There is an implicit assumption in the LTOP analyses that the accumulators are isolated when accumulator pressure is greater than the PORV lift setting in that there is no evaluation performed to demonstrate that a PORV can maintain RCS pressure below the LTOP limits should an accumulator inject into the RCS at a pressure greater than the PORV lift setting. Therefore, it is necessary to ensure that an accumulator does not inject in these circumstances. This assurance is provided by requiring the accumulators to be isolated and power be removed from the isolation valve operators. This ensures that a single event cannot result in the injection of an accumulator. RAIs 3.4-10 3.4-11 R1

   The LCO is changed to require that the accumulator isolation valves be closed and power removed from the isolation valve operators. The requirement to remove power from the isolation valves provides additional assurance that inadvertent accumulator injection does not occur. LCO 3.4.12, Action C, is revised to provide Conditions for an accumulator not isolated and for power available to an accumulator isolation valve. The Required Actions have been changed to require removing power from the affected isolation valve within one hour.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.12 Low Temperature Overpressure Protection (LTOP) System

3.4-11 ITS SR 3.4.12.3
STS SR 3.4.12.3
CTS 3.4.9.3
JFD 6

NRC RAI: JFD 6 states that the North Anna LTOP design assumes that an accumulator does not inject into the RCS while in the LTOP regime. ITS SR 3.4.12.3 has been modified by a NOTE to reflect the North Anna LTOP design. However, CTS 3.4.9.3 does not provide restrictions on the accumulator during LTOP applicability. UFSAR Section 5.2.2.2 does not discuss this assumption. **Comment:** Provide more information on the North Anna LTOP design assumptions and analyses.

Response: CTS 3.4.9.3 does not provide restrictions on the accumulators during the LTOP applicability because the model specification in NUREG-0452, Revision 3 (Westinghouse Standard Technical Specifications), which was the basis for the current North Anna specifications, did not include controls on accumulators in the LTOP specification. STS 3.4.12, LTOP System, applies restrictions on the accumulators. Therefore, in converting the North Anna CTS to the North Anna ITS, it was necessary to determine the appropriate controls on the accumulators in the LTOP regime. As stated in UFSAR Section 5.2.2.2, "Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analyses." There is an implicit assumption in the LTOP analyses that the accumulators are isolated when accumulator pressure is greater than the PORV lift setting in that there is no evaluation performed to demonstrate that a PORV can maintain RCS pressure below the LTOP limits should an accumulator inject into the RCS at a pressure greater than the PORV lift setting. Therefore, it is necessary to ensure that an accumulator does not inject in these circumstances. This assurance is provided by SR 3.4.12.3 requiring verification every 12 hours that the accumulators are isolated and power is removed from the isolation valve operators whenever accumulator pressure is greater than the PORV lift setting. This ensures that a single event cannot result in the injection of an accumulator. JFD 6 was expanded to include additional justification.

1. The North Anna LTOP system does not assume the operation of the RHR suction relief valves. References to the RHR suction relief valves are eliminated and the general term "RCS relief valves" is replaced with the more accurate "PORVs" throughout. The North Anna LTOP analysis assumes that only one charging pump and one Low Head Safety Injection (LHSI) pump are available for injection below the LTOP arming temperature. Appropriate changes are made to the ISTS and subsequent items are renumbered or relabeled as necessary.

2. North Anna Power Station is not adopting a Pressure Temperature Limits Report (PTLR) and is retaining the LTOP in the Technical Specifications. References to the PTLR have been deleted.

3. The brackets are removed and the proper plant specific information/value is provided.

4. Changes are made to reflect those changes made to the ISTS.

5. A requirement is added for verification that the PORV keyswitch is in the AUTO position in order to ensure that the LTOP System is activated. This Surveillance and Frequency are consistent with the CTS.

6. The North Anna LTOP design assumes that an accumulator does not inject into the RCS while in the LTOP protection regime. The North Anna CTS does not provide restrictions on the accumulator during LTOP applicability as the existing standard when the North Anna specifications were developed, NUREG-0452, did not address the accumulators in the LTOP range. Therefore, the accumulators were isolated under administrative controls. NUREG-1431 addresses the accumulators in the ISTS 3.4.12. Therefore, appropriate accumulator controls have been added to the North Anna ITS to reflect the North Anna design. RAIs 3.4-10 3.4-11 R1

   There is an implicit assumption in the LTOP analyses that the accumulators are isolated when accumulator pressure is greater than the PORV lift setting in that there is no evaluation performed to demonstrate that a PORV can maintain RCS pressure below the LTOP limits should an accumulator inject into the RCS at a pressure greater than the PORV lift setting. Therefore, it is necessary to ensure that an accumulator does not inject in these circumstances. This assurance is provided by requiring the accumulators to be isolated and power be removed from the isolation valve operators. This ensures that a single event cannot result in the injection of an accumulator. RAIs 3.4-10 3.4-11 R1

   The LCO is changed to require that the accumulator isolation valves be closed and power removed from the isolation valve operators. The requirement to remove power from the isolation valves provides additional assurance that inadvertent accumulator injection does not occur. LCO 3.4.12, Action C, is revised to provide Conditions for an accumulator not isolated and for power available to an accumulator isolation valve. The Required Actions have been changed to require removing power from the affected isolation valve within one hour.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.12 Low Temperature Overpressure Protection (LTOP) System

3.4-12 ITS LCO 3.4.12
STS LCO 3.4.12
CTS LCO 3.4.9.3
JFD 7
TSTF-280 Rev. 1

NRC RAI: JFD 7 notes that TSTF-280 Rev. 1 was not fully adopted due to changes made to the LCO to reflect the North Anna analysis and design. However, the only apparent TSTF-280 Rev. 1 change not adopted is "one of the following pressure relief capabilities:" which replaces "either a or b below." Attachment 3 of the North Anna submittal indicates that TSTF-280 Rev. 1 was incorporated. **Comment:** TSTF-280 Rev. 1 should be incorporated in its entirety.

Response: The Company will take the action proposed in the comment. The TSTF-280, Rev. 1 change of the LCO wording to "one of the following pressure relief capabilities:" will be adopted.

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

RAI 3.4-12 R1

LCO 3.5.3, Note #

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of [one] ~~[high pressure injection (HPI)] pump [and~~ one charging pump ① and one low head safety injection (LHSI) pump capable of injecting into the RCS and the accumulators isolated, with power removed from the isolation valve operators, and ~~either a or b below.~~ } ① TSTF-280 ①

LCO 3.4.9.3

~~a. Two RCS relief valves, as follows:~~ One of the following pressure relief capabilities:

~~1.~~ a. Two power operated relief valves (PORVs) with lift settings ~~within the limits specified in the PTLR,~~ of:

1. ≤ 500 psig (Unit 1), 415 psig (Unit 2) when any RCS cold leg temperature ≤ 235°F (Unit 1), 270°F (Unit 2)

2. ≤ 395 psig (Unit 1), 375 psig (Unit 2) when any RCS cold leg temperature ≤ 150°F (Unit 1), 130°F (Unit 2)

~~or~~ or ② ⑦ RAI 3.4-9 R1

~~[2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ [436.5] psig and ≤ [463.5] psig, or]~~ ①

~~[3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ [436.5] psig and ≤ [463.5] psig].~~ TSTF-285

b. The RCS depressurized and an RCS vent of ≥ ~~[2.07]~~ square inches. ③ TSTF-243

Applicability 3.4.9.3

APPLICABILITY: MODE 4 when ~~all~~ any RCS cold leg temperature is ≤ ~~[275]~~ 235°F (Unit 1), 270°F (Unit 2) ③
MODE 5,
MODE 6 when the reactor vessel head is on. ⑥

1. Two charging pumps may be made capable of injecting for ≤ 1 hour for pump swap operations.

Applicability 3.4.9.3

-----NOTE ⑤-----

② Accumulator isolation is only required when accumulator pressure is greater than or equal to the ~~maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.~~ the PORV lift setting ⑥ TSTF-285

With power removed from the isolation valve operators

Rev. 1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of one charging pump and one low head safety injection (LHSI) pump capable of injecting into the RCS and the accumulators isolated, with power removed from the isolation valve operators, and one of the following pressure relief capabilities: RAI 3.4-12 R1

a. Two power operated relief valves (PORVs) with lift settings of:

1. ≤ 500 psig (Unit 1), 415 psig (Unit 2) when any RCS cold leg temperature ≤ 235°F (Unit 1), 270°F (Unit 2)

2. ≤ 395 psig (Unit 1), 375 psig (Unit 2) when any RCS cold leg temperature ≤ 150°F (Unit 1), 130°F (Unit 2)

b. The RCS depressurized and an RCS vent of ≥ 2.07 square inches.

-------------------------------- NOTES --------------------------------

1. Two charging pumps may be made capable of injecting for ≤ 1 hour for pump swapping operations.

2. Accumulator isolation with power removed from the isolation valve operators is only required when accumulator pressure is greater than the PORV lift setting.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ 235°F (Unit 1), 270°F (Unit 2),
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two LHSI pumps capable of injecting into the RCS.	A.1 Initiate action to verify a maximum of one LHSI pump is capable of injecting into the RCS.	Immediately

3.4.17 RCS Loop Isolation Valves

3.4-13 ITS SR 3.4.17.1 and SR 3.4.17.2
STS SR 3.4.17.1
CTS SR 4.4.1.2
JFD 1

NRC RAI: ITS SR 3.4.17.1 and SR 3.4.17.2 proposes to split STS SR 3.4.17.1 into two surveillance requirements. This change appears to be generic in nature. **Comment:** A TSTF traveler should be submitted to generically change the STS to split SR 3.4.17.1 into two SRs. These changes also affect the ITS SR 3.4.17.1 Bases.

Response: The Company does not agree with the action recommended in the Comment. ITS SR 3.4.17.1 was split into two Surveillances because at North Anna (and Surry), there is no remote indication of loop isolation valve position after power is removed from the valve actuator. This is a plant-specific design feature. Other Westinghouse plants that have loop isolation valves and that have converted to ITS, such as Byron and Braidwood, adopted SR 3.4.17.1 as written in the STS. Therefore, this change is not generic.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.1 RCS Pressure, Temperature, and Flow DNB Limits

3.4-14 ITS LCO 3.4.1 BASES
STS LCO 3.4.1 BASES
JFD 1

NRC RAI: JFD 1 proposes to not include two paragraphs from the STS LCO 3.4.1 BASES. These paragraphs provide plant-specific information on measurement error on RCS flow rate and fouling. The JFD did not provide adequate information as to why these two paragraphs were not included into the ITS LCO 3.4.1 BASES. **Comment:** Provide justification for not including these two paragraphs or incorporate into ITS BASES.

Response: The Company will take the action proposed in the Comment. Additional justification is provided. As stated in JFD 1, changes have been made to the Bases to be consistent with the North Anna analysis and design. The discussion regarding inclusion of RCS total flow rate measurement error and bias to accommodate any undetected venturi fouling when performing the precision heat balance is not applicable to North Anna. North Anna utilizes a statistical combination of uncertainties when modeling DNBR which combines normal, steady state values with a statistical combination of uncertainties, such as measurement error. Therefore, there is no measurement error included in the RCS precision heat balance result.

BASES

Insert 1

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion ~~of ≥ [1.3]. This is the acceptance limit for the RCS DNB parameters.~~ Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.~~7~~6, "Control Bank Insertion Limits"; ~~LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4 "QUADRANT POWER TILT RATIO (QPTR)."~~ specified in the COLR 3 TSTF-136 TSTF-339

The pressurizer pressure limit ~~of [2200] psig~~ and ~~the~~ RCS average temperature limit ~~of [581]°F correspond to~~ equal the analytical limits ~~of [2205] psig and [596]°F used in the safety analyses, with allowance for measurement uncertainty.~~ because of the application of statistical combination of uncertainty. 2 1

The RCS DNB parameters satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). 3

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses~~.~~ Insert 2 ~~Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.~~ TSTF-339

~~RCS total flow rate contains a measurement error of [2.0]% based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of [0.1]% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to [2.1]% for no fouling.~~ 6 RAI 3.4-14 R1

~~Any fouling that might bias the flow rate measurement greater than [0.1]% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.~~

(continued)

Rev. 1

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.1 BASES, RCS PRESSURE, TEMPERATURE, AND FLOW DNB LIMITS

1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference 10 CFR 50.36.

4. The Applicability Bases are revised to reflect the plant-specific analyses. The discussion of non-applicable MODES is taken from the Bases of ITS 3.2.2.

5. The Bases are revised to reflect changes made to the ITS.

6. Changes have been made to the LCO Bases to be consistent with the North Anna analysis and design. The ISTS Bases discussion regarding inclusion of RCS total flow rate measurement error and bias to accommodate any undetected venturi fouling when performing the precision heat balance is not applicable to North Anna. North Anna utilizes a statistical combination of uncertainties when modeling DNBR which combines normal, steady state values with a statistical combination of uncertainties, such as measurement error. Therefore, there is no measurement error included in the RCS precision heat balance result. RAI 3.4-14 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.1 RCS Pressure, Temperature, and Flow DNB Limits

3.4-15 ITS SR 3.4.1.4 BASES
STS SR 3.4.1.4 BASES
JFD 5

NRC RAI: ITS SR 3.4.1.4 BASES does not include the discussion about the SR NOTE. This note was not incorporated into ITS SR 3.4.1.4. As stated in question 3.4-01, this note should be incorporated into ITS SR 3.4.1.4. **Comment:** Incorporated discussion of SR NOTE into ITS SR 3.4.1.4 BASES.

Response: The Company will take the action proposed in the Comment, with certain modifications. STS 3.4.1.4 requires verification by precision heat balance that RCS total flow rate is ≥ 295,000 gpm and greater than or equal to the limit specified in the COLR every 18 months. The STS SR is modified by a Note that states, "Not required to be performed until 24 hours after ≥ 90% RTP." This Note performs two purposes. First, it allows entry into MODE 1 if the Surveillance has not been met. Without the Note, if the Surveillance has not been met within its Frequency (18 months + 25%, or 22.5 months), entry into MODE 1 without first performing the Surveillance is prohibited by SR 3.0.4. As stated in JFD 2, this test is currently performed after each startup, typically a few weeks after achieving full power. The second purpose of the Note is to specify a maximum time after reaching full power following refueling before performing the Surveillance. The Note allows 24 hours after exceeding 90% RTP. As described in JFD 2, this restriction cannot be met at North Anna. Establishment of the conditions for performance of the precision heat balance is time consuming and requires installation of equipment and establishment of stable operating conditions. Twenty-four hours after exceeding 90% RTP does not allow sufficient time to establish stable plant conditions, install the instrumentation, perform the test, and analyze the results. Therefore, the Note has been modified to state, "Not required to be performed until 30 days after ≥ 90% RTP." The Bases have been modified as necessary to reflect this change. This provides time to reach a stable operating condition after startup, install the necessary equipment, perform the test, and analyze the results. See also the response to Question 3.4-1.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

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

SR 3.4.1.3

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

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every [18] months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. 2

The Frequency of [18] months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance. 2

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until ~~24 hours~~ 30 days after ≥ [90%] RTP. ~~This exception is appropriate since the heat balance requires the plant to be at a minimum of [90%] RTP to obtain the stated RCS flow accuracies.~~ The 30 day period after reaching 90% RTP is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. The Surveillance shall be performed within ~~24 hours~~ 30 days after reaching [90%] RTP. 2 5 3 RAI 3.4-15 A

REFERENCES

1. U FSAR, ~~Section~~ Chapter [15]. 1 2

Rev. 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 30 days after ≥ 90% RTP. The 30 day period after reaching 90% RTP is reasonable to establish stable operating conditions, install the test equipment, perform the test, and analyze the results. The Surveillance shall be performed within 30 days after reaching 90% RTP. RAI 3.4-15 R1

REFERENCES

1. UFSAR, Chapter 15.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.2 Minimum Temperature for Criticality

3.4-16 ITS SR 3.4.2.1 BASES

NRC RAI: Insert for ITS SR 3.4.2.1 BASES is missing an ‘s’ on Surveillance. The sentence should read ... and is consistent with other routine Surveillances which are typically performed once per shift. **Comment:** Add ‘s’ to Surveillance.

Response: The Company will take the action proposed in the Comment. The wording of the SR will be modified to state "Surveillances."

INSERT

RCS loop average temperature is required to be verified at or above 541 °F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

RAI 3.4-16 R1

BASES

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $k_{eff} < 1.0$ in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 541°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. RAI 3.4-16 R1 In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

REFERENCES

None.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.5 RCS Loops - MODE 3

3.4-17 ITS 3.4.5 ACTIONS C1, C2 and C3 BASES

NRC RAI: The BASES for ITS 3.4.5 ACTIONS C1, C2, and C3 were modified to delete the words "must be suspended." With this deletion, the BASES are not consistent with the ITS ACTIONS and STS BASES. **Comment:** The words "must be suspended" need to be reinserted into the ITS 3.4.5 ACTIONS C1, C2, and C3 BASES.

Response: The Company will take the action proposed in the Comment. The wording of the Action will be modified to restore the phrase, "must be suspended."

BASES

ACTIONS

C.1 and C.2 (continued)

the Required Action is either to restore the required RCS loop to operation or to de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets. When the RTBs are in the closed position and Rod Control System capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the RTBs must be opened. The Completion Times of 1 hour to restore the required RCS loop to operation or de-energize all CRDMs is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

(2)

C.1, C.2, and C.3 ~~D.1, D.2, and D.3~~ required a required not TSTF-263 (5) (1)

If (two) RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations ~~involving a reduction of RCS boron concentration~~ must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

place the Rod Control System in a condition incapable of rod withdrawal (e.g.)

TSTF-87

Insert 1

Insert 2

RAI 3.4-17 R1

TSTF-286

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required loops are in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

(continued)

Rev. 1

BASES

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing unit conditions in an orderly manner and without challenging unit systems.

C.1, C.2, and C.3

If two required RCS loops are inoperable or a required RCS loop is not in operation, except as during conditions permitted by the Note in the LCO section, place the Rod Control System in a condition incapable of rod withdrawal (e.g., all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets). All operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation. RAI 3.4-17 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.5 RCS Loops - MODE 3

3.4-18 ITS SR 3.4.5.3 BASES
TSTF-265 Rev. 2

NRC RAI: STS SR 3.4.5.3 requires the licensee to verify correct breaker alignment and indicated power are available to each required pump. This wording was approved via TSTF-265 Rev. 2. ITS SR 3.4.5.3 BASES did not adopt TSTF-265 Rev. 2 in its entirety. ITS SR 3.4.5.3 would verify correct breaker alignment and indicated power to the required pump not in operation. The TSTF revised the SR to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating.
Comment: TSTF-265 Rev. 2 should be adopted in its entirety.

Response: The Company does not agree with the action recommended in the Comment. TSTF-265 revises SR 3.4.8.2 from, "Verify correct breaker alignment and indicated power are available to the required pump that is not in operation," to "Verify correct breaker alignment and indicated power are available to each required pump." The TSTF-265 modifications to the SR would require performance of SR 3.4.8.2 on operating loops when, as stated in the TSTF-265 changes to the Bases, operation is evidence of proper breaker alignment and power availability. As a result, the proposed changes in TSTF-265 add additional administrative burden with no compensatory increase in safety. Therefore, the Company will retain the CTS requirements and will propose a generic change to the ISTS. Attachment 3 of the submittal has been revised to indicate that TSTF-265 was only partially incorporated. See also the response to Questions 3, 4, 5, 6, 21, 24, and 26.

3.4.6 RCS Loops - MODE 4

3.4-19 ITS 3.4.6 APPLICABILITY BASES

NRC RAI: ITS 3.4.6 APPLICABILITY BASES proposed to change the BASES wording from "meet single failure considerations" to "provide redundancy for heat removal." The JFD number beside the proposed change is 6. However, JFD 6 does not exist for this section. **Comment:** Provide JFD for proposed change.

Response: The Company will take the action proposed in the Comment. JFD 6 was inadvertently deleted and will be restored. JFD 6 states, "The Applicability Bases state, 'However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.' In the Background section of the Bases for this Specification, the need for a second loop is stated as, 'The other intent of this LCO is to require that two paths be OPERABLE to provide redundancy for heat removal.' This is a more accurate statement of the requirement. The term 'single failure' is typically used to describe an accident analysis assumption and the accident analyses performed for MODE 4 do not assume the single failure of a heat removal loop. The Applicability Bases have been revised to describe the requirement using the wording from the Bases Background section."

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.6 BASES, RCS LOOPS - MODE 4

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

3. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference 10 CFR 50.36.

4. The Bases are changed to state that two decay heat removal paths must be OPERABLE instead of stating that two loops must be available. The LCO requires two paths to be OPERABLE. The term "available" is unclear in this context and could lead to misinterpretation of the requirements.

5. Changes are made to reflect those changes made to the ISTS. The following requirements are renumbered or revised, where applicable, to reflect the changes.

6. The Applicability Bases state, "However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations." In the Background section of the Bases for this Specification, the need for a second loop is stated as, "The other intent of this LCO is to require that two paths be OPERABLE to provide redundancy for heat removal." This is a more accurate statement of the requirement. The term "single failure" is typically used to describe an accident analysis assumption and the accident analyses performed for MODE 4 do not assume the single failure of a heat removal loop. The Applicability Bases have been revised to describe the requirement using the wording from the Bases Background section. RAI 3.4-19 R1

3.4.6 RCS Loops - MODE 4

3.4-20 ITS 3.4.6 ACTIONS B1 and B2 BASES
TSTF-263 Rev. 3

NRC RAI: ITS 3.4.6 ACTIONS B1 and B2 BASES proposed word changes to the BASES. It is stated that the proposed changes are consistent with TSTF-263. Some of the proposed changes are not consistent with TSTF-263 and no JFD was provided. **Comment:** TSTF-263 wording should be retained, otherwise provide justification for proposed changes. If the proposed changes are generic, then a TSTF traveler should be proposed.

Response: The Company will take the action proposed in the Comment. TSTF-263, Revision 1, was incorporated instead of the approved Revision 3. The Bases will be revised to incorporate TSTF-263, Revision 3.

BASES

ACTIONS

~~B.1~~ A.2 (continued)

~~loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.~~

If restoration is not accomplished and an RHR loop is OPERABLE.

If ~~the parameters that are outside the limits cannot be restored~~, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 ~~(≤ 200°F)~~ rather than MODE 4 ~~(200 to 300°F)~~. The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging ~~plant~~ unit systems.

Insert 1

TSTF-263 ②

~~C~~B.1 and ~~C~~B.2

two required ⑤ are inoperable a required loop is not

If ~~no~~ loop ~~is OPERABLE~~ or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving ~~a reduction of RCS boron concentration~~ must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. ~~Boron dilution requires forced circulation for proper mixing, and the~~ margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

required Insert 2 Insert 3

TSTF-263 RAI 3.4-20 R1 TSTF-286

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

TSTF-263

This SR requires verification every 12 hours that ~~one~~ the required RCS or RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

(continued)

Rev. 1

BASES

ACTIONS (continued)

A.2

If restoration is not accomplished and an RHR loop is OPERABLE, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging unit systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if an RHR loop is OPERABLE. With no RHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on the restoration of an RHR loop, rather than a cooldown of extended duration.

B.1 and B.2

If two required loops are inoperable or a required loop is not in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation. RAI 3.4-20 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.6 RCS Loops - MODE 4

3.4-21 ITS SR 3.4.6.3 BASES
TSTF-265 Rev. 2

NRC RAI: STS SR 3.4.6.3 requires the licensee to verify correct breaker alignment and indicated power are available to each required pump. This wording was approved via TSTF-265 Rev. 2. ITS SR 3.4.6.3 BASES did not adopt TSTF-265 Rev. 2 in its entirety. ITS SR 3.4.6.3 would verify correct breaker alignment and indicated power to the required pump not in operation. The TSTF revised the SR to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating.
Comment: TSTF-265 Rev. 2 should be adopted in its entirety.

Response: The Company does not agree with the action recommended in the Comment. TSTF-265 revises SR 3.4.8.2 from, "Verify correct breaker alignment and indicated power are available to the required pump that is not in operation," to "Verify correct breaker alignment and indicated power are available to each required pump." The TSTF-265 modifications to the SR would require performance of SR 3.4.8.2 on operating loops when, as stated in the TSTF-265 changes to the Bases, operation is evidence of proper breaker alignment and power availability. As a result, the proposed changes in TSTF-265 add additional administrative burden with no compensatory increase in safety. Therefore, the Company will retain the CTS requirements and will propose a generic change to the ISTS. Attachment 3 of the submittal has been revised to indicate that TSTF-265 was only partially incorporated. See also the response to Questions 3, 4, 5, 6, 18, 24, and 26.

3.4.7 RCS Loops - MODE 5, Loops Filled

3.4-22 ITS 3.4.7 ACTIONS C1 and C2 BASES
TSTF-263 Rev.3

NRC RAI: ITS 3.4.7 ACTIONS C1 and C2 BASES proposed word changes to the BASES. It is stated that the proposed changes are consistent with TSTF-263. However, not all of the changes as described in TSTF-263 Rev. 3 were incorporated. Specifically, the first part of the first sentence should state "If a required RHR loop is not in operation,..." ITS BASES state "If no required RHR loop is in operation,..." **Comment:** TSTF-263 Rev. 3 wording should be retained, otherwise provide justification for proposed changes.

Response: The Company will take the action proposed in the Comment. TSTF-263, Revision 3 will be incorporated. This also resulted in editorial changes ITS 3.4.7, Conditions A and B.

BASES

APPLICABILITY (continued)

or the secondary side water level of at least ~~[two]~~ one SG~~s~~ is required to be ≥ ~~[~~17~~]~~%. with the associated loop isolation valves open

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3
5

If all RCS loops are isolated, an SG cannot be used for decay heat removal and RCS water inventory is substantially reduced. In this circumstances LCO 3.4.8 applies.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

1

ACTIONS

A.1 ~~and~~ A.2 B.1, and B.2

3

If one RHR loop is ~~inoperable~~ OPERABLE and ~~the~~ any required SG~~s~~ ~~have~~ has secondary side water level~~s~~ < ~~[~~17~~]~~%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water level~~s~~. Either Required Action ~~A.1 or Required Action A.2~~ will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

1

TSTF-263

C.1 and C.2

4

a

RAI 3.4-22 R1

If ~~no~~ a required RHR loop is not in operation, except during conditions permitted by Note 1 and Note 4, or if no required loop is OPERABLE, all operations involving ~~a reduction of RCS boron concentration~~ Insert 1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. ~~To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation.~~ Insert 2 The immediate Completion Times reflect the importance of maintaining operation for heat removal.

TSTF-286

(continued)

Rev. 0

BASES

APPLICABILITY

In MODE 5 with the unisolated portion of the RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SG is required to be ≥ 17% with the associated loop isolation valves open.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

If all RCS loops are isolated, an SG cannot be used for decay heat removal and RCS water inventory is substantially reduced. In this circumstance, LCO 3.4.8 applies.

ACTIONS

A.1, A.2, B.1, and B.2

If one RHR loop is OPERABLE and any required SG has secondary side water level < 17%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water level. Either Required Action will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

C.1 and C.2

If a required RHR loop is not in operation, except during conditions permitted by Note 1 and Note 4, or if no required RHR loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of RAI 3.4-22 R1

(continued)

INSERT

A.	One required RHR loop inoperable. AND One RHR loop OPERABLE.	A.1	Initiate action to restore a second RHR loop to OPERABLE status.	Immediately	
		OR			
		A.2	Initiate action to restore required SGs secondary side water level to within limits.	Immediately	RAI 3.4-22 R1
B.	One or more required SGs with secondary side water level not within limits. AND One RHR loop OPERABLE.	B.1	Initiate action to restore a second RHR loop to OPERABLE status.	Immediately	RAI 3.4-22 R1
		OR			
		B.2	Initiate action to restore required SGs secondary side water level to within limits.	Immediately	RAI 3.4-22 R1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RHR loop inoperable. AND One RHR loop OPERABLE.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SGs secondary side water level to within limits.	Immediately Immediately
B. One or more required SGs with secondary side water level not within limits. AND One RHR loop OPERABLE.	B.1 Initiate action to restore a second RHR loop to OPERABLE status. OR B.2 Initiate action to restore required SGs secondary side water level to within limits.	Immediately Immediately
C. No required RHR loops OPERABLE. OR Required RHR loop not in operation.	C.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meed SDM of LCO 3.1.1. AND C.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

RAI 3.4-22 R1
RAI 3.4-22 R1
RAI 3.4-22 R1
RAI 3.4-22 R1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.7 RCS Loops - MODE 5, Loops Filled

3.4-23 ITS 3.4.7 ACTIONS C1 and C2
TSTF-263 Rev. 3

NRC RAI: ITS 3.4.7 ACTIONS C1 and C2 BASES proposed word changes to the BASES. It is stated that the proposed changes are consistent with TSTF-263. Some of the proposed changes are not consistent with TSTF-263. The JFD (JFD 4) that was provided was not sufficient to justify the proposed changes. **Comment:** TSTF-263 wording should be retained, otherwise provide justification for proposed changes. If the proposed changes are generic, then a TSTF traveler should be proposed.

Response: The Company will take the action proposed in the Comment. JFD 4 is expanded to describe the changes made to the STS that are not associated with TSTF-263.

JUSTIFICATION FOR DEVIATIONS
ITS 3.4.7 BASES, RCS LOOPS - MODE 5, LOOPS FILLED

1. Changes are made to reflect those changes made to the ISTS. The following requirements are renumbered or revised, where applicable, to reflect the changes.

2. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference 10 CFR 50.36.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. The ISTS Bases to 3.4.7, Conditions C.1 and C.2 (as modified by TSTF-263) state, "If a required RHR loop is not in operation, except during conditions permitted by Note 1". However, both Note 1 and Note 4 allow the required RHR loop to not be in operation. Therefore, the Bases are revised to state, "except during conditions permitted by Note 1 and Note 4." RAI 3.4-23 R1

5. Information was added to the Bases to clarify the effect of the loop isolation valves on the Specification. The LCO Bases are modified to make clear that the loop isolation valves associated with any Steam Generator being credited to meet the LCO requirements must be open. In addition, clarification is added to the Applicability portion of the Bases to make clear that the condition "loops filled" applies to those portions of the RCS that are not isolated from the RCS. This appropriate because the steam generators in unisolated loops can still be used as a heat sink.

6. The LCO Bases state, "An additional RHR loop is required to be OPERABLE to meet single failure considerations." In the Background section of the Bases for this Specification, the need for a second RHR loop is stated as, "The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal." This is a more accurate statement of the requirement. The term "single failure" is typically used to describe an accident analysis assumption and the accident analyses performed for MODE 5 do not assume the single failure of an RHR loop. The LCO Bases have been revised to describe the LCO requirement using the wording from the Bases Background section.

7. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.7 RCS Loops - MODE 5, Loops Filled

3.4-24 ITS SR 3.4.7.3 BASES
TSTF-265 Rev. 2

NRC RAI: STS SR 3.4.7.3 requires the licensee to verify correct breaker alignment and indicated power are available to each required pump. This wording was approved via TSTF-265 Rev. 2. ITS SR 3.4.7.3 BASES did not adopt TSTF-265 Rev. 2 in its entirety. ITS SR 3.4.7.3 would verify correct breaker alignment and indicated power to the required pump not in operation. The TSTF revised the SR to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating.
Comment: TSTF-265 Rev. 2 should be adopted in its entirety.

Response: The Company does not agree with the action recommended in the Comment. TSTF-265 revises SR 3.4.8.2 from, "Verify correct breaker alignment and indicated power are available to the required pump that is not in operation," to "Verify correct breaker alignment and indicated power are available to each required pump." The TSTF-265 modifications to the SR would require performance of SR 3.4.8.2 on operating loops when, as stated in the TSTF-265 changes to the Bases, operation is evidence of proper breaker alignment and power availability. As a result, the proposed changes in TSTF-265 add additional administrative burden with no compensatory increase in safety. Therefore, the Company will retain the CTS requirements and will propose a generic change to the ISTS. Attachment 3 of the submittal has been revised to indicate that TSTF-265 was only partially incorporated. See also the response to Questions 3, 4, 5, 6, 18, 21 and 26.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.8 RCS Loops - MODE 5, Loops Not Filled

3.4-25 ITS 3.4.8 ACTIONS B1 and B2 BASES
TSTF-263 Rev. 3

NRC RAI: ITS 3.4.8 ACTIONS B1 and B2 BASES proposed word changes to the BASES. It is stated that the proposed changes are consistent with TSTF-263. Some of the proposed changes are not consistent with TSTF-263 and no JFD was provided. **Comment:** TSTF-263 wording should be retained, otherwise provide justification for proposed changes. If the proposed changes are generic, then a TSTF traveler should be proposed.

Response: The Company will take the action proposed in the Comment. TSTF-263, Revision 1, was incorporated instead of the approved Revision 3. The Bases will be revised to incorporate TSTF-263, Revision 3. This also resulted in a editorial change to ITS 3.4.8, Condition B markup (changed "loops" to "loop".) The typed ITS is correct.

BASES

ACTIONS (continued)

B.1 and B.2

If no required ~~RHR~~ loops ~~are~~ is OPERABLE or the required loop is not in operation, except during conditions permitted by Note 1, all operations involving ~~a reduction of RCS boron concentration~~ Insert 1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. ~~Boron dilution requires forced circulation for uniform dilution, and~~ The required margin to criticality must not be reduced in this type of operation. Insert 2 The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

TSTF-263 | RAI 3.4-25 R1

TSTF-286

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that ~~one~~ the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

TSTF-263

SR 3.4.8.2

Verification that the required ~~number of~~ pumps ~~are~~ is OPERABLE ~~ensures that~~ an additional pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states that the SR is not required to be performed until 24 hours after a required pump is not in operation.

TSTF-265 6

REFERENCES

None.

Rev.1

BASES

ACTIONS (continued)

B.1 and B.2

If no required loop is OPERABLE or the required loop is not in operation, except during conditions permitted by Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

RAI 3.4-25 R1

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.8.2

Verification that the required pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.

CTS

RAI 3.4-25 R1

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action b	B. ~~Required~~ No RHR loops ~~inoperable~~ OPERABLE. OR Required ~~No~~ RHR loop not in operation.	B.1 ~~Suspend all operations involving reduction in RCS boron concentration.~~ AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately	TSTF-286 TSTF-263

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.4.1.3.4.b	SR 3.4.8.1	Verify ~~one~~ required RHR loop is in operation.	12 hours	TSTF-263
4.4.1.3.2	SR 3.4.8.2	------NOTE------ Not required to be performed until 24 hours after a required pump is not in operation. ------------ Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days	TSTF-265 2

Rev. 1

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.8 RCS Loops - MODE 5, Loops Not Filled

3.4-26 ITS SR 3.4.8.2 BASES
TSTF-265 Rev. 2

NRC RAI: STS SR 3.4.8.2 requires the licensee to verify correct breaker alignment and indicated power are available to each required pump. This wording was approved via TSTF-265 Rev. 2. ITS SR 3.4.8.2 BASES did not adopt TSTF-265 Rev. 2 in its entirety. ITS SR 3.4.8.2 would verify correct breaker alignment and indicated power to the required pump not in operation. The TSTF revised the SR to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating.
Comment: TSTF-265 Rev. 2 should be adopted in its entirety.

Response: The Company does not agree with the action recommended in the Comment. TSTF-265 revises SR 3.4.8.2 from, "Verify correct breaker alignment and indicated power are available to the required pump that is not in operation," to "Verify correct breaker alignment and indicated power are available to each required pump." The TSTF-265 modifications to the SR would require performance of SR 3.4.8.2 on operating loops when, as stated in the TSTF-265 changes to the Bases, operation is evidence of proper breaker alignment and power availability. As a result, the proposed changes in TSTF-265 add additional administrative burden with no compensatory increase in safety. Therefore, the Company will retain the CTS requirements and will propose a generic change to the ISTS. Attachment 3 of the submittal has been revised to indicate that TSTF-265 was only partially incorporated. See also the response to Questions 3, 4, 5, 6, 18, 21 and 24.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.11 Pressurizer Power Operated Relief Valves

3.4-27 ITS 3.4.11 ACTIONS D1 and D2 BASES
STS 3.4.11 ACTIONS C1 and C2 BASES

NRC RAI: The proposed wording changes to the ITS 3.4.11 ACTIONS D1 and D2 BASES (mark up copy) is not consistent with the STS and ITS BASES. The mark up copy of ITS states "... 72 hours, the PORV may be returned to manual control." The STS BASES states "... 72 hours, the power will be restored to the PORV." The clean copy of the ITS BASES states "... 72 hours, the PORV may be returned to automatic control." No JFD was provided for the changes.
Comment: Provide justification for the correct proposed change or retain the STS BASES.

Response: The Company will take the action proposed in the Comment. TSTF-151 and WOG-ED-20, revised the ITS 3.4.11 Action D.1 Bases to state, "If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation." This wording will be adopted in the North Anna ITS. WOG-ED-20 also modified the 3.4.11 LCO Bases. This has also been incorporated.

BASES

ACTIONS

D.1 and D.2 (continued) (8)

status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition C, since the PORVs ~~are not~~ May not be capable of mitigating an overpressure event ~~when placed in manual control~~ if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, ~~the power will be restored and the PORV restored to OPERABLE status~~ the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the unit must be brought to a MODE in which the LCO does not apply, as required by Condition E. (B) TSTF-151 WOG-ED-20 (1) RAI 3.4-27 R1 (8) TSTF-309

Insert

E.1 and E.2 (8)

If the Required Action of Condition A, B, C, or D is not met, then the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4 ~~and 5~~ automatic PORV OPERABILITY ~~may be~~ is required. See LCO 3.4.12. (1) (4) TSTF-151

F.1, F.2, ~~E.3, and E.4~~ (8) (9)

If more than one PORV is inoperable and not capable of being manually cycled, ~~it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time~~ (9)

(continued)

Rev. 1

BASES

ACTIONS (continued)

D.1 and D.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition C, since the PORVs may not be capable of mitigating an event if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the unit must be brought to a MODE in which the LCO does not apply, as required by Condition E. RAI 3.4-27 R1

The Required Actions D.1 and D.2 are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with another Required Action. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition C entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s).)

E.1 and E.2

If the Required Action of Condition A, B, C, or D is not met, then the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within

(continued)

INSERT 1

As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function.

INSERT 2

An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage.) Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

WOG-EO-20
RAI 3.4-27
R1

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage to within LCO 3.4.13, "RCS Operational Leakage." 1

BASES

BACKGROUND (continued)	Pressure–High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
APPLICABLE SAFETY ANALYSES	Unit operators employ the PORVs to depressurize the RCS in response to certain unit transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator. The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical (Ref. 2). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function. Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR. By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV RAI 3.4-27 R1

(continued)

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.11 Pressurizer Power Operated Relief Valves

3.4-28 ITS 3.4.11 ACTIONS E1 and E2 BASES
STS 3.4.11 ACTIONS E1 and E2 BASES

NRC RAI: STS 3.4.11 ACTIONS E1 and E2 require that the associated block valve be closed and power removed from the associated block valve if two PORVs are inoperable and not capable of being manually cycled. ITS 3.4.11 did not incorporate these action items. CTS 3.4.3.2 ACTION A5 has the same STS 3.4.11 ACTIONS E1 and E2 requirements. **Comment:** STS 3.4.11 ACTIONS E1 and E2 BASES should be incorporated into ITS 3.4.11 BASES.

Response: The Company does not agree with the action recommended in the Comment. ITS 3.4.11, Condition C, states that with one PORV inoperable and not capable of being manually cycled, close the associated block valve, remove power from the associated block valve, and restore the PORV to OPERABLE status. The ACTIONS are modified by a Note stating that, "Separate Condition entry is allowed for each PORV and each block valve." ITS 3.4.11, Condition F, applies with "Two PORVs inoperable and not capable of being manually cycled." Under the ITS rules of multiple condition entry (see Example 1.3-5 in ITS 1.3), when Condition F is entered for two PORVs inoperable, Condition C is also entered concurrently for each inoperable valve. Therefore, the Condition C Required Actions C.1 and C.2 are duplicative of the STS ACTIONS E.1 and E.2 (what would be STS Required Actions F.1 and F.2). As these duplicative Required Actions are unnecessary and confusing, they are removed. In the CTS these actions are not duplicative, as the CTS does not allow multiple condition entry. In the CTS, Action A.4 for one inoperable PORV is exited and Action A.5 is entered when two PORVs become inoperable. Therefore, these Actions are required in both CTS Actions A.4 and A.5. See also the response to Question 3.4-07.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.15 RCS Leakage Detection Instrumentation

3.4-29 ITS SR 3.4.15.2 BASES

NRC RAI: ITS SR 3.4.15.2 BASES proposed to change the wording of the last sentence which differs from the STS. The proposed wording states that "the Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 3)." **Comment:** A TSTF traveler should be submitted to generically change the STS BASES in section SR 3.4.15.2.

Response: The Company does not agree with the action recommended in the Comment. The STS Bases state, "The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation." This sentence is not applicable to North Anna and was revised. North Anna currently performs this Surveillance monthly. Therefore, there is no operating experience at North Anna which demonstrates that the ITS 92 day Frequency is proper for detecting degradation. The Bases were revised to provide a justification for the SR Frequency that is accurate for North Anna. The STS Bases are accurate for any plant that currently has an SR Frequency of 92 days, so the change is not generic.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

3.4.16 RCS Specific Activity

3.4-30 ITS 3.4.16 BASES JFDs

NRC RAI: The list of JFDs for the ITS 3.4.16 BASES lists JFD 5 and JFD 6. However, these JFDs do not appear in the ITS 3.4.16 BASES mark up. **Comment:** Specify where these changes occur.

Response: The Company will take the action proposed in the Comment, with certain modifications. JFD 5 and 6 do not apply to 3.4.16 and will be deleted.

1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Criterion 4 describes systems which are important contributors to risk. Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference the appropriate 10 CFR 50.36 Criterion.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. The Reference to 10 CFR 100.11 is revised to eliminate the referenced year. The most recent version of the Code of Federal Regulations is applicable and referencing a year is unnecessary.

5. Not used. RAI 3.4-30

6. Not used. R1

3.4.15 RCS Leakage Detection Instrumentation

3.4-31 CTS LCO 3.4.6.1
ITS LCO 3.4.15
ITS 3.4.15 BACKGROUND BASES

NRC RAI: CTS requires three diverse methods of leakage detection to be operable in Modes 1, 2, 3, and 4. ITS LCO 3.4.15 only requires two diverse methods of leakage detection. The proposed insert 1 to ITS 3.4.15 BASES states that the "UFSAR Chapter 3 (Ref. 1) requires compliance with Regulatory Guide 1.45." However, the regulatory position of Reg Guide 1.45 states that "at least three separate detection methods should be employed." The proposed change from CTS LCO 3.4.6.1 to ITS LCO 3.4.15 is not consistent with the guidance in Reg Guide 1.45 or the proposed ITS BASES. **Comment:** The current licensing basis as specified in the CTS should be maintained.

Response: The Company will take the action proposed in the Comment, with certain modifications. As stated in the NRC Safety Evaluation for North Anna Amendments 118 / 102, dated July 7, 1989, the CTS requires two separate and independent leakage detection systems: the containment atmosphere particulate and gaseous radioactivity monitoring system and the containment sump level and discharge flow measurement system. The Safety Evaluation states that the containment particulate radiation monitor and gaseous radiation monitor are not considered as two independent systems because they share a common piping system, power supply, and pump arrangement. The Safety Evaluation states that the CTS meets the recommendations of Regulatory Guide 1.45 (Regulatory Position 9 regarding Technical Specifications). The North Anna ITS also will require two separate and independent leakage detection systems: the containment atmosphere radioactivity monitoring system (particulate or gaseous) and the containment sump monitor (level or discharge flow). Therefore, the current licensing basis as specified in the CTS is maintained in the ITS. Two diverse measurement means are required by the ITS, but the Bases state that requiring only one instrument of each diverse method provides an acceptable minimum. This is consistent with Regulatory Guide 1.45, Position 9.

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," describes acceptable methods of implementing General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary." As such, Regulatory Guide 1.45 describes the design of leakage detection systems. The quoted statement, "at least three separate detection methods should be employed," is taken from Regulatory Position 3, which, as stated in the introductory paragraph of Section C, "Regulatory Position," describes how the "leakage detection and collection systems should be selected and designed." The North Anna design is consistent with Regulatory Guide 1.45, as described in UFSAR Section 5.2.4.1.1.

Regulatory Guide 1.45, Section C, Item 9, addresses Technical Specifications, and states, "The technical specifications should include the limiting conditions for identified and unidentified leakage and address the availability of various types of instruments to assure adequate coverage at all times." ITS LCO 3.4.15 is consistent with Regulatory Guide 1.45, Regulatory Position 9 regarding Technical Specifications. There are no statements in the ITS Bases that are in conflict with Regulatory Guide 1.45.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

STS 3.4.15 states, "The following RCS leakage detection instrumentation shall be OPERABLE: a. One containment sump (level or discharge flow) monitor, b. One containment atmosphere radioactivity monitor (gaseous or particulate), and [c. One containment air cooler condensate flow rate monitor.]" Under the NUREG conventions, requirements in brackets are omitted if they do not match the plant design. Since North Anna does not have containment air cooler condensate flow rate monitors, the proper implementation of LCO 3.4.15 would be, "The following RCS leakage detection instrumentation shall be OPERABLE: a. One containment sump (level or discharge flow) monitor, and b. One containment atmosphere radioactivity monitor (gaseous or particulate)." This plant-specific application of the STS is consistent with other plant designs which do not have containment air cooler flow rate monitoring. For example, the Byron and Braidwood CTS was worded similarly to the North Anna CTS and the approved ITS for Byron and Braidwood stations are identical to that proposed for North Anna.

The Company believes the LCO should be adopted as proposed. The North Anna proposed LCO wording is consistent with the application of the STS to our plant-specific design, is identical to what has been approved by the NRC for other, similarly designed, Westinghouse plants, and is consistent with Position 9 of Regulatory Guide 1.45.

North Anna Improved Technical Specifications (ITS) Review Comments
ITS Section 3.4, Reactor Coolant System

CHANGES NOT ASSOCIATED WITH RAI RESPONSES

1. TSTF-367, Rev. 0, is incorporated. Note that the changes in TSTF-367 had been incorporated into some specifications and only minor changes to the ISTS markup was needed in most cases.

2. The TSTF-61, Rev. 0, insert into SR 3.4.13.1 is revised to capitalize the defined term "LEAKAGE."

3. Editorial change NRC-ED-7 is incorporated into 3.4.16, Condition A, replacing a plant-specific editorial change.

4. The insert to the Applicable Safety Analyses Bases for 3.4.16 on ISTS page B 3.4-94 is revised to refer to a radiologically limiting "SGTR" instead of "STGR."

5. The ACTIONS Notes for 3.4.11 are revised to state "NOTES" instead of "NOTE." The label in the ISTS markup is correct.

6. The ACTIONS Notes for 3.4.14 are revised to state "NOTES" instead of "NOTE." The label in the ISTS markup is correct.

7. The CTS 3.4.10.1 Discussions of Change page header is corrected.

8. The Note to 3.4.11, Required Actions D.1 and D.2 is corrected to be the full width of the column.

9. The Completion Times of 3.4.11, Required Actions F.1 and F.2 are corrected from 1 hour to 6 hours and 12 hours, respectively. The STS markup is correct.

10. The Completion Time of 3.4.11, Required Action G.1 is moved down to be aligned with the Required Action.

11. The Notes to 3.4.15, Required Action A.1 and Required Action B.1.2 are corrected to be only the width of the associated Required Action, not the full column width.

12. A typographical error is corrected in 3.4.7, Required Action C.1, and 3.4.8, Required Action B.1. The word "meed" is changed to "meet."

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The Remote Shutdown System is considered an important contributor to the reduction of unit risk to accidents and as such it has been retained in the Technical Specifications as indicated in the NRC Policy Statement.~~ satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

3 TSTF 367 R1

LCO

The Remote Shutdown System LCO provides the OPERABILITY requirements of the instrumentation and controls necessary to ~~place and~~ maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls ~~typically~~ required are listed in Table 3.3.4-1 ~~in the accompanying LCO~~. B

1 4 TSTF 266

~~Reviewer's Note: For channels that fulfill GDC 19 requirements, the number of OPERABLE channels required depends upon the unit licensing basis as described in the NRC unit specific Safety Evaluation Report (SER). Generally, two divisions are required OPERABLE. However, only one channel per a given Function is required if the unit has justified such a design, and NRC's SER accepted the justification.~~ 5

The controls, instrumentation, and transfer switches are required for:

- Core reactivity control (~~initial and~~ long term); 1
- RCS pressure control;
- Decay heat removal via the AFW System and the SG ~~safety valves or SG ADVs;~~ PORVs 2
- RCS inventory control via charging flow~~; and~~ 4
- ~~Safety support systems for the above Functions, including service water, component cooling water, and onsite power, including the diesel generators.~~ 2

A Function of a Remote Shutdown System is OPERABLE if all instrument and control channels needed to support the Remote Shutdown System Function are OPERABLE. In some cases, B Table 3.3.4-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long TSTF-266

(continued)

Rev. 1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops—MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

three

The reactor coolant is circulated through ~~[four]~~ RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification. (1)

OPERABLE

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be ~~available~~ to provide redundancy for decay heat removal. (4)

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops—MODE 4 ~~have been identified in the NRC Policy Statement as important contributors to risk reduction.~~ (3)

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS

(continued)

Satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

TSTF-367 | R1

Rev. 0

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops—MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through three RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be OPERABLE to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops—MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). R1

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

RCS Loops—MODE 5 (Loops Filled) ~~have been identified in the NRC Policy Statement as important contributors to risk reduction.~~ TSTF-367 (2)

(7) using narrow range instrumentation

(5) LCO

and the associated loop isolation valves open

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or ~~two~~ SGs with secondary side water level ≥ [17]%. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE ~~to meet single failure considerations~~. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is ~~two~~ SGs with ~~their~~ secondary side water levels ≥ [17]%. Should the operating RHR loop fail, the SGs could be used to remove the decay heat. (a) (1) (3)

TSTF-114 its

via natural circulation

(a) using narrow range instrumentation (7) (1) (3)

to provide redundancy for heat removal (6)

not in operation

TSTF-153

pump swap operations or (1)

Note 1 permits all RHR pumps to be ~~de-energized~~ ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits ~~de-energizing~~ of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

TSTF-153 stopping

swap pumps or (7)

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

(continued)

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops—MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. ~~The number of pumps in operation can vary to suit the operational needs.~~ The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal. 5

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) ~~have been identified in the NRC Policy Statement as important contributors to risk reduction.~~ satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). TSTF-367 1 R1

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to ~~meet single failure considerations~~ provide redundancy for heat removal. 3

(continued)

Rev. 1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops—MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). R1

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RHR pumps to not be in operation for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS)~~, as required by GDC 34,~~ to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System. ①

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit ~~de-energizing~~ loop to not be in operation the RHR ~~pump~~ for short durations, under the condition that the boron concentration is not diluted. This conditional ~~de-energizing~~ removal from operation of the RHR ~~pump~~ loop does not result in a challenge to the fission product barrier. TSTF-153

~~Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk~~ satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii). ② TSTF-367 R1

(continued)

Rev. 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~reduction. Therefore, the RHR System is retained as a Specification.~~

TSTF-367 R1

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the ~~low end~~ RHR discharge (3) temperature. The flow path starts in one of the RCS hot legs and is returned to at least one of the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to ~~be removed from service~~ not be in operation for up to 1 hour per 8 hour period, provided no operations are permitted that would ~~cause a reduction of~~ dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

TSTF (3) 153

TSTF 286

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft

(continued)

Rev. 1

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit the RHR loop to not be in operation for short durations, under the condition that the boron concentration is not diluted. This conditional removal from operation of the RHR loop does not result in a challenge to the fission product barrier.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). R1

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), ~~as required by GDC 34,~~ to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System. ①

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

~~Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.~~ ②

Satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE.

(continued)

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B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

The RHR System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii). R1

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

(continued)

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 4.4.6.2.1.d	SR 3.4.13.1 ------------------NOTE------------------ Not required to be performed ~~in MODE 3 or 4~~ until 12 hours after establishment of steady state operation. ------------------------------------ Verify RCS operational LEAKAGE is within limits by performance of ~~Perform~~ RCS water inventory balance.	~~-----NOTE----- Only required to be performed during steady state operation -----------~~ TSTF-116 72 hours
SR 4.4.5.0	SR 3.4.13.2 Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

TSTF-61

R1

Rev. D

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	------------------NOTE------------------ Not required to be performed until 12 hours after establishment of steady state operation. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours R1
SR 3.4.13.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

3.4.8

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
New (Note) Table 4.4-4, item 4a) MODE 1-5, - Action A Action a	A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	------------Note------------ LCO 3.0.4 is not applicable. ---------------------------- A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1. AND A.2 Restore DOSE EQUIVALENT I-131 to within limit.	Once per 4 hours 48 hours	NRC-EO-7 NRC-EO-7
Action b	B. Gross specific activity of the reactor coolant not within limit.	~~B.1 Perform SR 3.4.16.2.~~ ~~AND~~ B.~~2~~ 1 Be in MODE 3 with T_{avg} < 500°F.	~~4 hours~~ 6 hours	TSTF-28

(continued)

Rev. 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

release rate to

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 ~~activity in~~ the reactor coolant by a factor of ~~about 50~~ immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē μCi/gm for gross specific activity.

500

30①

radiologically limiting SGTR

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

1① R1

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. ~~The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.~~

①

①

LCO

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR

(continued)

Rev. 1

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 release rate to the reactor coolant by a factor of 500 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The radiologically limiting SGTR analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. R1

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours.

The remainder of the above LCO limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTES------------------------------------ R1

1. Separate Condition entry is allowed for each PORV and each block valve.

2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable due to inoperable backup nitrogen supply and capable of being manually cycled.	A.1 Restore backup nitrogen supply to OPERABLE status.	14 days
B. One or more PORVs inoperable for reason other than Condition A and capable of being manually cycled.	B.1 Close and maintain power to associated block valve.	1 hour
C. One PORV inoperable and not capable of being manually cycled.	C.1 Close associated block valve.	1 hour
	AND	
	C.2 Remove power from associated block valve.	1 hour
	AND	
	C.3 Restore PORV to OPERABLE status.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV required to be tested shall be within limit.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except any required valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation.

ACTIONS

------------------------------------ NOTES ------------------------------------ R1

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more required RCS PIVs not within limit.	A.1 Restore RCS PIV leakage to within limit.	4 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

 R1

ADMINISTRATIVE CHANGES

None

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

R.1 CTS 3.4.10.1 provides requirements for the ASME Code Class 1, 2 and 3 components to ensure their structural integrity. These requirements are in addition to the requirements in CTS 4.0.5. This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3.4.10.1 does not meet the 10 CFR 50.92(c)(2)(ii) criteria for inclusion into the ITS.

10 CFR 50.36(c)(2)(ii) Criteria Evaluation:

1. The ASME Code Class 1, 2 & 3 Components requirements are not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. The ASME Code Class 1, 2 & 3 Components inspection requirements do not satisfy criterion 1.

2. The ASME Code Class 1, 2 & 3 Components requirements are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The ASME Code Class 1, 2 & 3 Components inspection requirements do not satisfy criterion 2.

3. The ASME Code Class 1, 2 & 3 Components requirements are not a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The ASME Code Class 1, 2 & 3 Components inspection requirements do not satisfy criterion 3.

4. The ASME Code Class 1, 2 & 3 Components requirements are not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. As discussed in Section 4.0, (Appendix A, page A-43) of WCAP-11618, the

R1

ASME Code Class 1, 2 & 3 Components requirements were found to be a non-significant risk contributor to core damage frequency and offsite releases. The Company has reviewed this evaluation, considers it applicable to the North Anna Power Station, and concurs with this assessment. The requirements in this Specification are not important for any scenarios modeled in the North Anna Power Station site-specific PRAs. The ASME Code Class 1, 2 & 3 Components inspection requirements do not meet criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the ASME Code Class 1, 2 & 3 Components LCO and associated Applicability, and Actions may be relocated out of the Technical Specifications. The ASME Code Class 1, 2 & 3 Components specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the LCO did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

None

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One block valve inoperable.	-------------NOTE------------ Required Action D.1 and D.2 do not apply when block valve is inoperable solely as a result of complying with Required Action C.2. ----------------------------	R1
	D.1 Place associated PORV in manual control.	1 hour
	AND	
	D.2 Restore block valve to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3.	6 hours
	AND	
	E.2 Be in MODE 4.	12 hours
F. Two PORVs inoperable and not capable of being manually cycled.	F.1 Be in MODE 3.	6 hours R1
	AND	
	F.2 Be in MODE 4.	12 hours R1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Two block valves inoperable.	G.1 ---------NOTE--------- Required Action G.1 does not apply when block valve is inoperable solely as a result of complying with Required Action C.2. ---------------------- Restore one block valve to OPERABLE status.	 2 hours R1
H. Required Action and associated Completion Time of Condition G not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	Verify PORV backup nitrogen supply pressure is within limit.	7 days
SR 3.4.11.2	-------------------NOTES------------------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each block valve.	 92 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump (level or discharge flow) monitor; and

b. One containment atmosphere radioactivity monitor (gaseous or particulate).

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

ACTIONS

------------------- NOTE -------------------
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1 ---------NOTE--------- Not required until 12 hours after establishment of steady state operation. ----------------------	R1
	Perform SR 3.4.13.1.	Once per 24 hours R1
	AND	
	A.2 Restore required containment sump monitor to OPERABLE status.	30 days

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1	Analyze grab samples of the containment atmosphere.	Once per 24 hours
	OR		
	B.1.2	---------NOTE--------- Not required until 12 hours after establishment of steady state operation. ---------------------	
		Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	B.2	Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. All required monitors inoperable.	D.1	Enter LCO 3.0.3.	Immediately

North Anna Power Station
Units 1 and 2
Improved TS Review Comments
ITS Section 3.6, Containment Systems

3.6.1 Containment

1. Discussion of Changes (DOC) A.8 (CTS 1.0)
(3.6.1-1) CTS 1.6
CTS 3/4.6
ITS 3.6.1, 3.6.2, 3.6.3, and Associated Bases

NRC RAI: CTS 1.6 defines CONTAINMENT INTEGRITY. A markup of CTS 1.6 is provided in the CTS markup of CTS 1.0, but not in the markup of CTS 3.6. DOC A.8 (CTS 1.0) states that the definition of CONTAINMENT INTEGRITY is deleted from the CTS/ITS. This is not entirely correct. The DOC is incorrect in that the definition is not deleted but is relocated to various Bases in ITS 3.6, which is a Less Restrictive (LA) change. In addition, there are Administrative changes associated with CTS 1.6, which deal with the requirements of the definition being used as the basis for certain SRs in ITS 3.6.1, 3.6.2 and 3.6.3. CTS 1.6, Item 1.6.1 is the basis for ITS SRs 3.6.3.1, 3.6.3.2, 3.6.3.3, and 3.6.3.4; Item 1.6.3 is the basis for ITS 3.6.2, and Item 1.6.4 is the basis for ITS SRs 3.6.1.1 and 3.6.1.2. Refer to Comment Numbers 3.6.1-2 and 3.6.1-3.
Comment: Revise the CTS markup and provide the appropriate discussions and justifications for these Administrative and Less Restrictive (LA) changes.

Response: The Company will take the action proposed in the Comment.

CTS 1.6.1 is marked as part of ITS 3.6.3 adopting the requirement using DOC A.1. Requirements for CTS 1.6.1 are included as being related to ITS SR 3.6.3.1, SR 3.6.3.2, SR 3.6.3.3, and 3.6.3.4.

CTS 1.6.3 is remarked as part of ITS 3.6.2 adopting the requirement using DOC A.1. Requirements for CTS 1.6.3 have been marked as part of ITS 3.6.2.

CTS 1.6.4 is remarked as part of ITS SR 3.6.1.1 adopting the requirement using DOC A.1. ISTS 3.6.1.2 is not adopted.

An LA DOC is not used because the material is retained in the ITS, not moved to another document. DOC A.1 is used instead.

CTS Pages in Section 1.0 are marked to describe to which ITS sections the respective requirements are being moved.

ITS 3.6.1

A.1

5-5-83

ITS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, expressed in % of RATED THERMAL POWER between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

See ITS 1.0

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

LCO 3.6.1 1.6 CONTAINMENT ~~INTEGRITY~~ shall exist when: OPERABILITY A.4

1.6.1 All penetrations required to be closed during accident conditions are either:

See ITS 3.6.3

RAI 3.6.3-1 R1

RAI 3.6.1-1 R1

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Rev 1

ITS 3.6.1

A.1

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

ITS

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

3.6.1 1.6.2 All equipment hatches are closed and sealed. L.1

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. LA.1

SR 3.6.1.1 1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.2>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

See ITS 1.0

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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ITS 3.6.1

ITS

5-5-83

1.0 DEFINITIONS

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1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

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CONTAINMENT INTEGRITY

LCO 3.6.1

1.6 CONTAINMENT INTEGRITY shall exist when: OPERABILITY

A.4

1.6.1 All penetrations required to be closed during accident conditions are either:

See ITS 3.6.3

RAI 3.6.3-1 R1

RAI 3.6.1-1 R1

REV 1

A.1

ITS 3.6.1

ITS

RAI 3.6.1-1 RAI 3.6.2-1 RAI 3.6.3-1 R1

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b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

3.6.1 1.6.2 All equipment hatches are closed and sealed L.1

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. LA.1

<See ITS 3.6.2>

SR 3.6.1.1 1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

RAI 3.6.1-3 RAI 3.6.1-2 R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

See ITS 1.0

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ITS

A.1

ITS 3.6.2

RAI 3.6.1-1
RAI 3.6.1-2
RAI 3.6.1-3
R1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

1.6.2 All equipment hatches are closed and sealed.

See ITS 3.6.1

3.6.2 1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3.

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and

See ITS 3.6.1

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.

See ITS 3.6.3

RAI 3.6.1-3
RAI 3.6.1-2
R1

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1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

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ITS 3.6.2

ITS

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

1.6.2 All equipment hatches are closed and sealed. See ITS 3.6.1

3.6.2 1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3.

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and See ITS 3.6.1

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. See ITS 3.6.3

RAI 3.6.1-3
RAI 3.6.1-2
R1

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(A.1)

ITS 3.6.3

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1.0 DEFINITIONS

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See ITS 1.0

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b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when: (See ITS 3.6.1)

SR 3.6.3.1
SR 3.6.3.2
SR 3.6.3.3
SR 3.6.3.4

1.6.1 All penetrations required to be closed during accident conditions are either:

RAI 3.6.3-1 R1

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ITS 3.6.3

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

SR 3.6.3.1
SR 3.6.3.2
SR 3.6.3.3
SR 3.6.3.4

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

SR 3.6.3 ACTIONS Note

1.6.2 All equipment hatches are closed and sealed.

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. <See ITS 3.6.1>

<See ITS 3.6.2>

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.1>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. L.14

RAI 3.6.1-3

RAI 3.6.1-2

R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

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<See ITS 1.0>

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

A.1

ITS 3.6.3

ITS

5-5-83

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

<See ITS 1.0>

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, expressed in % of RATED THERMAL POWER between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when: <See ITS 3.6.1>

SR 3.6.3.1
SR 3.6.3.2
SR 3.6.3.3
SR 3.6.3.4

1.6.1 All penetrations required to be closed during accident conditions are either:

RAI 3.6.3-1 R1

A.1

ITS 3.6.3

ITS

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

SR 3.6.3.1
SR 3.6.3.2
SR 3.6.3.3
SR 3.6.3.4
SR 3.6.3 ACTIONS Note

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.6.2 All equipment hatches are closed and sealed. <See ITS 3.6.1>

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. <See ITS 3.6.2>

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.1>

1.6.5 ~~The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.~~ L.14

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

<See ITS 1.0>

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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ITS Chapter 1.0

A.1

1.0 USE AND APPLICATION

5-5-83

ITS Section 1.1

1.~~0~~ 1 DEFINITIONS

NOTE: The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases. A.1

ACTION S

1.1 ACTION S shall be that part of a Specification which prescribes ~~remedial~~ Required Actions to be taken ~~measures required~~ under designated conditions within specified Completion Times. A.1

Add proposed definition of Actuation Logic Test A.2

AXIAL FLUX DIFFERENCE

AFD 1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, ~~expressed in % of RATED THERMAL POWER~~ between the top and bottom halves of a two section excore neutron detector. A.1

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY ~~the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST.~~ The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping Insert 1 or total channel steps ~~such that the entire channel is calibrated~~. A.12

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and~~/or~~ status with other indications ~~and~~/or status derived from independent instrumentation channels measuring the same parameter. A.1

CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST (COT) L.1

COT 1.5 A ~~CHANNEL FUNCTIONAL TEST~~ shall be~~:~~ A.1

~~a. Analog channels -~~ the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY ~~including alarm and/or trip functions~~ Insert 2 of all devices in the channel required for channel OPERABILITY. A.11

~~b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.~~ A.4

A.8

~~CONTAINMENT INTEGRITY~~

~~1.6 CONTAINMENT INTEGRITY shall exist when:~~ <See ITS 3.6.1> RAI 3.6.1-1 R1

~~1.6.1 All penetrations required to be closed during accident conditions are either:~~ <See ITS 3.6.3> RAI 3.6.3-1 RAI 3.6.1-1 R1

A.1

ITS Chapter 1.0

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

ITS
Section 1.1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

1.6.2 All equipment hatches are closed and sealed. See ITS 3.6.1

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. See ITS 3.6.2

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and See ITS 3.6.1

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. See ITS 3.6.3

R1 RAI 3.6.1-3
R1 RAI 3.6.1-2

~~CONTROLLED LEAKAGE~~

~~1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~ A.9

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement ~~or manipulation~~ of any ~~component~~ fuel, sources, or reactivity control components within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe ~~conservative~~ position. L.2 A.5

CORE OPERATING LIMITS REPORT (COLR)

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides ~~core operating~~ Cycle specific parameter limits for the current ~~operating~~ reload cycle. These cycle-specific ~~core operating~~ parameter limits shall be determined for each reload cycle in accordance with Specification ~~6.9.1.7~~ 5.6.5. Plant operation within these ~~operating~~ limits is addressed in individual specifications. A.1

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites". A.1

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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A.1

1.0 USE AND APPLICATION

5-5-83

1.~~0~~ 1 DEFINITIONS

ITS Section 1.1

NOTE: The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases. A.1

~~ACTION~~ (S)

1.1 ACTION (S) shall be that part of a Specification which prescribes Required Actions to be taken ~~remedial measures required~~ under designated conditions within specified Completion Times. A.1

Add proposed definition of Actuation Logic Test A.2

AXIAL FLUX DIFFERENCE AFD

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, ~~expressed in % of RATED THERMAL POWER~~ between the top and bottom halves of a two section excore neutron detector. A.1

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY ~~the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST~~. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps ~~such that the entire channel is calibrated~~. Insert 1 A.12

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and~~/or~~ status with other indications ~~and/~~or status derived from independent instrumentation channels measuring the same parameter. A.1

CHANNEL ~~FUNCTIONAL~~ OPERATIONAL TEST (COT)

1.5 A ~~CHANNEL FUNCTIONAL TEST~~ COT shall be~~:~~

~~a. Analog channels -~~ the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY ~~including alarm and/or trip functions~~ of all devices in the channel required for channel OPERABILITY. Insert 2 L.1 A.1 A.11

~~b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.~~ A.4 A.8

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

1.6.1 All penetrations required to be closed during accident conditions are either:

<See ITS 3.6.1> RAI 3.6.1-1 R1

<See ITS 3.6.3> RAI 3.6.1-1 RAI 3.6.3-1 R1

A.1

ITS Chapter 1.0

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

ITS Section 1.1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

<See ITS 3.6.3>

1.6.2 All equipment hatches are closed and sealed. <See ITS 3.6.1>

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. <See ITS 3.6.2>

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.1>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

R1 RAI 3.6.1-3

R1 RAI 3.6.1-2

~~CONTROLLED LEAKAGE~~

~~1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~ A.9

CORE ALTERATION

fuel, sources, or reactivity control components L.2

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position. A.5

CORE OPERATING LIMITS REPORT (COLR)

parameter

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications. A.1

cycle specific parameter

5.6.5

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites". A.1

Ē-AVERAGE DISINTEGRATION ENERGY

1.11 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

3.6.1 Containment

2. DOC A.8 (CTS 1.0)
(3.6.1-2) CTS 1.6.2
ITS B3.6.1 Bases - BACKGROUND

NRC RAI: CTS 1.6 defines CONTAINMENT INTEGRITY. A markup of CTS 1.6 is provided in the CTS markup of CTS 1.0, but not in the markup of CTS 3.6. DOC A.8 (CTS 1.0) states that the definition of CONTAINMENT INTEGRITY is deleted from the CTS/ITS. This justification is incorrect. CTS 1.6.2 states that "All equipment hatches are closed and sealed." ITS B3.6.1 Bases - BACKGROUND states the following: "To maintain this leak tight barrier:. All equipment hatches are closed." The requirement for sealing the equipment hatches has been deleted. No justification is provided for this Less Restrictive (L) change. **Comment:** Provide a discussion and justification for this Less Restrictive (L) change.

Response: The Company will take the action proposed in the Comment. CTS 1.6.2 is remarked as part of ITS 3.6.1. The reference to the equipment hatches being closed is moved to the Bases and justified by DOC LA.1. The reference to the equipment hatches being sealed is deleted and justified by DOC L.1. CTS Pages in Section 1.0 are marked to describe to which ITS sections the respective requirements are being moved.

ITS 3.6.1

A.1

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
- R1

ITS

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

3.6.1 1.6.2 All equipment hatches are closed and sealed. L.1

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3 LA.1

SR 3.6.1.1 1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.2>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

See ITS 1.0

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

(A.1)

ITS 3.6.1

ITS

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

3.6.1 1.6.2 All equipment hatches are closed and sealed L.1

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. LA.1

See ITS 3.6.2

SR 3.6.1.1 1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. See ITS 3.6.3

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

See ITS 1.0

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

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A.1

ITS 3.6.2
RAI 3.6.1-1
RAI 3.6.1-2
RAI 3.6.1-3
R1

ITS

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

1.6.2 All equipment hatches are closed and sealed.

3.6.2 1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. See ITS 3.6.1

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and See ITS 3.6.1

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. See ITS 3.6.3

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

See ITS 1.0

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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A.1

ITS ITS 3.6.2

RAI 3.6.1-1 RAI 3.6.2-1 RAI 3.6.3-1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

1.6.2 All equipment hatches are closed and sealed. See ITS 3.6.1

3.6.2 1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3.

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and See ITS 3.6.1

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. See ITS 3.6.3

RAI 3.6.1-3

RAI 3.6.1-2

R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

See ITS 1.0

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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Rev 1

(A.1)

ITS 3.6.3

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

SR 3.6.3.1
SR 3.6.3.2
SR 3.6.3.3
SR 3.6.3.4
SR 3.6.3 ACTIONS Note

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.6.2 All equipment hatches are closed and sealed. ⟨See ITS 3.6.1⟩

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. ⟨See ITS 3.6.2⟩

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and ⟨See ITS 3.6.1⟩

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. (L.14)

RAI 3.6.1-3

RAI 3.6.1-2

R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

⟨See ITS 1.0⟩

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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A.1

ITS 3.6.3

ITS

 RAI 3.6.1-1 RAI 3.6.2-1 RAI 3.6.3-1 R1

SR 3.6.3.1 SR 3.6.3.2 SR 3.6.3.3 SR 3.6.3.4 SR 3.6.3 ACTIONS Note

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.6.2 All equipment hatches are closed and sealed. <See ITS 3.6.1>

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. <See ITS 3.6.2>

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.1>

1.6.5 ~~The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.~~ L.14

RAI 3.6.1-3 RAI 3.6.1-2 R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

<See ITS 1.0>

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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The purpose of CTS 3.6.1.6 is to ensure action is taken expeditiously to restore containment structural integrity if it is not within limits. This change is acceptable because a 1 hour Completion Time is representative of the importance to take action expeditiously. Containment structural integrity problems once confirmed are unlikely to be corrected in as short a period of time as 1 or 24 hours. The 1 hour time frame is consistent with the ITS 3.0.3 requirement to make preparations to place the unit outside the MODE of Applicability. This change is considered more restrictive because the completion time for an action in the CTS is reduced.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 2 – Removing Descriptions of System Operation)* CTS 1.6 states, "CONTAINMENT INTEGRITY shall exist when:...1.6.2 All equipment hatches are closed and sealed." 3.6.1 states, "Containment shall be OPERABLE." This changes the CTS by moving the reference to the equipment hatch being closed to the Bases. The change deleting the phrase "and sealed" is addressed by DOC L.1. RAI 3.6.1-2 R1

The removal of these details, which are related to system operation, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains the requirement to perform required visual inspections and leakage rate testing in accordance with the Containment Leakage Rate Testing Program in accordance with 10 CFR 50 Appendix J, Part B, which would provide verification that the equipment hatch is closed. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system operation is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirements)* CTS 1.6 states, "CONTAINMENT INTEGRITY shall exist when:...1.6.2 All equipment hatches are closed and sealed." 3.6.3 states, "Each containment isolation valve shall be OPERABLE." This changes the CTS by not including an explicit reference to sealing the equipment hatches. The change associated with moving the reference to the equipment hatch to the Bases is addressed by DOC LA.1.

The purpose of CTS 1.6.2 is to help provide assurance that the equipment hatches can perform their safety function. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The Containment Leakage Rate Testing Program requires testing be performed in accordance with 10 CFR 50 Appendix J, Part B, requiring the containment isolation valves, including the equipment hatch, is OPERABLE, but there is no specific mention of sealing the equipment hatches. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

RAI
3.6.1-2
R1

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
RI

ITS Section 1.1

1.0 ① DEFINITIONS (Continued) 4-22-94

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

<See ITS 3.6.3>

1.6.2 All equipment hatches are closed and sealed. <See ITS 3.6.1>

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. <See ITS 3.6.2>

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.1>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

RI RAI 3.6.1-3

RI RAI 3.6.1-2

~~CONTROLLED LEAKAGE~~

~~1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~ A.9

CORE ALTERATION

fuel, sources, or reactivity control components L.2

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position. A.5

CORE OPERATING LIMITS REPORT (COLR)

parameter

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 5.6.5. Plant operation within these operating limits is addressed in individual specifications. A.1

Cycle specific parameter

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites". A.1

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

A.1

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

ITS Section 1.1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

<See ITS 3.6.3>

1.6.2 All equipment hatches are closed and sealed. <See ITS 3.6.1>

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. <See ITS 3.6.2>

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.1>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

R1 RAI 3.6.1-3

R1 RAI 3.6.1-2

~~CONTROLLED LEAKAGE~~

~~1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~ A.9

CORE ALTERATION

fuel, sources, or reactivity control components L.2

1.8 CORE ALTERATION shall be the movement ~~or manipulation~~ of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe ~~conservative~~ position. A.5

CORE OPERATING LIMITS REPORT (COLR)

parameter

Cycle specific parameter

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification ~~6.9.1.7~~ 5.6.5. Plant operation within these operating limits is addressed in individual specifications. A.1

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$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

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3.6.1 Containment

3. DOC A.8 (CTS 1.0)
(3.6.1-3) Bases JFD 2
CTS 1.6.5
STS B3.6.1 Bases - BACKGROUND
ITS B3.6.1 Bases - BACKGROUND

NRC RAI: CTS 1.6 defines CONTAINMENT INTEGRITY. A markup of CTS 1.6 is provided in the CTS markup of CTS 1.0, but not in the markup of CTS 3.6. DOC A.8 (CTS 1.0) states that the definition of CONTAINMENT INTEGRITY is deleted from the CTS/ITS. DOC A.8 is incorrect. CTS 1.6.5 states that "The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE." STS B3.6.1.1 Bases - BACKGROUND has a similar statement defining the leaktight barrier. ITS B3.6.1.1 Bases - BACKGROUND deletes this statement based on changes made to the ISTS (Bases JFD 2). Since CTS 1.6.5 is contained in the CTS and no changes to the ISTS were made with regards to this item, it needs to be included in ITS B3.6.1.1 Bases - BACKGROUND. **Comment:** Revise ITS B3.6.1.1 Bases - BACKGROUND to include CTS 1.6.5 or provide additional discussion and justification for its deletion based on system design, operational constraints, or current licensing basis.

Response: The Company will take the action proposed in the Comment. CTS 1.6.5 is marked as part of ITS 3.6.3. Requirements for CTS 1.6.5 are deleted and justified by DOC L.14. CTS Pages in Section 1.0 are marked to describe to which ITS sections the respective requirements are being moved.

ITS 3.6.1

A.1

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
- R1

ITS

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

3.6.1 1.6.2 All equipment hatches are closed and sealed. L.1

1.6.3 ~~Each air lock is OPERABLE pursuant to Specification 3.6.1.3~~ LA.1

SR 3.6.1.1 1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.2>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

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See ITS 1.0

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(A.1)

ITS 3.6.1

ITS

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

3.6.1 1.6.2 All equipment hatches are closed and sealed L.1

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. LA.1

See ITS 3.6.2

SR 3.6.1.1 1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. See ITS 3.6.3

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

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See ITS 1.0

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A.1

ITS 3.6.2

RAI 3.6.1-1
RAI 3.6.1-2
RAI 3.6.1-3
R1

ITS

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

See ITS 3.6.3

1.6.2 All equipment hatches are closed and sealed.

See ITS 3.6.1

3.6.2 1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3.

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and

See ITS 3.6.1

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.

See ITS 3.6.3

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

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CORE ALTERATION

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See ITS 1.0

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ITS 3.6.2

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

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See ITS 3.6.3

1.6.2 All equipment hatches are closed and sealed. See ITS 3.6.1

3.6.2 1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3.

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and See ITS 3.6.1

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. See ITS 3.6.3

RAI 3.6.1-3
RAI 3.6.1-2
R1

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

See ITS 1.0

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ITS 3.6.3

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

ITS

SR 3.6.3.1
SR 3.6.3.2
SR 3.6.3.3
SR 3.6.3.4

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SR 3.6.3 ACTIONS Note

1.6.2 All equipment hatches are closed and sealed.

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. ⟨See ITS 3.6.1⟩

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and ⟨See ITS 3.6.2⟩ ⟨See ITS 3.6.1⟩

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. L.14

RAI 3.6.1-3

RAI 3.6.1-2

R1

CONTROLLED LEAKAGE

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⟨See ITS 1.0⟩

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ITS 3.6.3

ITS

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

SR 3.6.3.1
SR 3.6.3.2
SR 3.6.3.3
SR 3.6.3.4
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1.6.5 ~~The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.~~ L.14

RAI 3.6.1-3
RAI 3.6.1-2
R1

<See ITS 1.0>

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1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

misinterpreting the requirements of the Surveillance Requirement while maintaining the assumptions of the accident analysis. This change is designated as less restrictive because less stringent Surveillance Requirements are being applied in the ITS than were applied in the CTS. RAI 3.6.3-21 R1

L.14 *(Category 1 – Relaxation of LCO Requirements)* CTS 1.6 states, "CONTAINMENT INTEGRITY shall exist when:...1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE." 3.6.3 states, "Each containment isolation valve shall be OPERABLE." This changes the CTS by not including an explicit reference to the sealing mechanisms associated with each penetration being OPERABLE. RAI 3.6.1-3 R1

The purpose of CTS 1.6.5 is to help provide assurance that the penetration isolation devices can perform their safety function. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The Containment Leakage Rate Testing Program requires testing be performed in accordance with 10 CFR 50 Appendix J, Part B, and each containment isolation valve and containment air lock is required to be OPERABLE, but there is no specific mention of the sealing mechanisms. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

<See ITS 3.6.3>

1.6.2 All equipment hatches are closed and sealed. <See ITS 3.6.1>

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. <See ITS 3.6.2>

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.1>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

R1 RAI 3.6.1-3
R1 RAI 3.6.1-2

~~CONTROLLED LEAKAGE~~

~~1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~ A.9

CORE ALTERATION

fuel, sources, or reactivity control components L.2

1.8 CORE ALTERATION shall be the movement ~~or manipulation~~ of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position. A.5

CORE OPERATING LIMITS REPORT (COLR)

parameter

Cycle specific parameter

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 (5.6.5). Plant operation within these operating limits is addressed in individual specifications. A.1

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites". A.1

Ē-AVERAGE DISINTEGRATION ENERGY

1.11 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

A.1

RAI 3.6.1-1
RAI 3.6.2-1
RAI 3.6.3-1
R1

ITS

Section 1.1

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

<See ITS 3.6.3>

1.6.2 All equipment hatches are closed and sealed. <See ITS 3.6.1>

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3. <See ITS 3.6.2>

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2 and <See ITS 3.6.1>

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE. <See ITS 3.6.3>

R1 RAI 3.6.1-3

R1 RAI 3.6.1-2

~~CONTROLLED LEAKAGE~~

~~1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~

A.9

CORE ALTERATION

fuel, sources, or reactivity control components

L.2

1.8 CORE ALTERATION shall be the movement ~~or manipulation~~ of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

A.5

CORE OPERATING LIMITS REPORT (COLR)

parameter

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7 Plant operation within these operating limits is addressed in individual specifications.

Cycle specific parameter

5.6.5

A.1

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites".

A.1

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

3.6.1 Containment

4. DOC A.1
(3.6.1-4) Bases JFD 3
CTS 4.6.1.1.c
CTS 4.6.1.1.d
CTS 3/4.6.1.2
ITS SR 3.6.1.1 and Associated Bases

NRC RAI: CTS 4.6.1.1.c, 4.6.1.1.d, and 4.6.1.2 require leak rate testing in accordance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix J, Option B. STS SR 3.6.1.1 requires the visual examination and leakage rate testing be performed in accordance with 10 CFR 50 Appendix J as modified by approved exemptions. ITS SR 3.6.1.1 modifies STS SR 3.6.1.1 to conform to CTS 4.6.1.2 as modified in the CTS markup. The STS is based on Appendix J, Option A while the CTS and ITS are based on Appendix J, Option B. Changes to the STS with regards to Option A versus Option B are covered by TSTF-52, Rev. 3. The changes to ITS 3.6.2 are in conformance with TSTF-52-Rev.3; however, ITS 3.6.1, 3.6.3, and the Bases for ITS 3.6.1 and 3.6.3 may not be in conformance with TSTF-52. Refer to Comment Numbers 3.6.1-5, 3.6.1-6, 3.6.2.7 and 3.6.3-2. **Comment:** Licensee should revise its submittal to conform to TSTF-52, Rev. 3.

Response: The Company will take the action proposed in the Comment. TSTF-52 Rev 3 includes markups for adopting 10 CFR 50 Appendix J, Option B, which is the version of TSTF-52 Rev 3 adopted in ITS 3.6.1.

Three changes were identified as not conforming to TSTF-52 without a JFD.

1. In the ITS 3.6.1 Applicable Safety Analysis Bases, TSTF-52 changed "loss of coolant accident (LOCA)" to "LOCA." The ISTS markup and typed ITS are revised to reflect this change.
2. In the SR 3.6.1.1 Bases, TSTF-52 did not change "< 0.6" to "≤ 0.6." JFD 7 is added to justify the deviation.
3. In the SR 3.6.2.1 Bases, the TSTF is corrected by changing "criteria which is" to "criteria which are," and JFD 10 is added to justify this change.

Retaining ISTS SR 3.6.3.7 includes the associated portion of TSTF-52.

BASES

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. All equipment hatches are closed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA, a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% of containment air weight per day in the safety analyses at P_a = 44.1 psig (Ref. 3). RAI 3.6.1-4 R1

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the applicable leakage limits must be met.

(continued)

BASES

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. All equipment hatches are closed; and

[d. The pressurized sealing mechanism associated with a penetration is OPERABLE, except as provided in LCO 3.6.[ ].] ③

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.1]% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be [0.1]% of containment air weight per day in the safety analyses at P_a = [40.4] 44.1 psig (Ref. 3).

TSTF-52 RAI 3.6.1-4 R1

③

TSTF-52

③ ⑥

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

(continued)

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BASES

ACTIONS

A.1 (continued)

also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 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. unit (1)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions. Failure to meet air lock [and purge valve with resilient seal] leakage limits specified in LCO 3.6.2 [and LCO 3.6.3] does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required 10 CFR 50 Appendix J leakage test is required to be ≤ 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

the Containment Leakage Rate Testing Program

Containment Leakage Rate Testing Program

TSTF-52 (3) (3) RAI 3.6.1-6 R1

(7) RAI 3.6.1-4 R1 TSTF-52

TSTF-52

INSERT A

(5) TSTF-52

(continued)

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with ~~the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.~~ the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by ~~Appendix J (Ref. 1), as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~ the Containment Leakage Rate Testing Program. TSTF-52 TSTF-52

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria ~~of~~ which ~~is~~ are applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the ~~overall~~ combined Type B and C containment leakage rate. 10 RAI 3.6.1-4 R1 TSTF-52 TSTF-52

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur when combined with administrative procedures. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is ~~only~~ not normally challenged when the containment air lock door is ~~opened~~ used for entry and exit (procedures require strict adherence to single door opening), ~~this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently~~ Insert 3 8 TSTF17

(continued)

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2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.7 4

2

RAI 3.6.3-2 3.6.1-4 3.6.1-5 R1

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. TSTF-52 Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3). 2 3

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened. 2 2

prior to entering MODE 4 from MODE 5 after containment vacuum has been broken. This

This Frequency will ensure that each time these valves are cycled they will be leak tested

SR 3.6.3.8 5

2

Automatic power operated containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 3 1 3

(continued)

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JUSTIFICATION FOR DEVIATIONS
ITS 3.6.1 BASES, CONTAINMENT

1. Changes are made (additions, deletions, and/or changes) to the ISTS which reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description.

2. Changes are made to reflect those changes made to the ISTS. The following requirements are renumbered or revised, where applicable, to reflect the changes.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. The criteria of the NRC Final Policy Statement on Technical Specifications Improvements have been included in 10 CFR 50.36(c)(2)(ii). Therefore, references in the ISTS Bases to the NRC Final Policy Statement are revised in the ITS Bases to reference 10 CFR 50.36.

5. Reviewer's note not retained.

6. Editorial change made for enhanced clarity or to be consistent with the ISTS Writers Guide.

7. For combined Type B and C tests, the leakage limit is $\leq 0.6\ L_a$ instead of $< 0.6\ L_a$ in the Bases of SR 3.6.1.1, consistent with technical changes made as part of TSTF-52. RAI 3.6.1-4 R1

3.6.1 Containment

5. DOC A.1
(3.6.1-5) DOC A.26 (CTS 6.0)
Bases JFD 3
CTS 4.6.1.1.c
CTS 4.6.1.1.d
STS SR 3.6.1.1, SR 3.6.2.1 and SR 3.6.3.7
ITS 3.6.1 and 3.6.3 and Associated Bases
ITS 5.5.15

NRC RAI: CTS 4.6.1.1.c and 4.6.1.1.d require specific leak rate tests for the containment equipment hatch and the butterfly isolation valves in the containment purge and the containment vacuum ejector lines. The CTS markup of CTS 4.6.1.1.c and 4.6.1.1.d in CTS 3.6 refers the reviewer to ITS 5.5.15 for changes associated with these specifications. The CTS markup for ITS 5.5.15 relocates these two specifications out of the ITS to the Containment Leakage Rate Testing Program. This change is justified by DOC A.26 (CTS 6.0). This change is incorrect. ITS 5.5.15 does not contain the specifics of these two specifications; the specifics are contained in the body of the program, which is outside of TS. Thus the change, if acceptable, would be a Less Restrictive (LA) change. However, the staff concludes that these two specifications may need to be retained in the North Anna ITS. Amendment 196 and 177 to the North Anna Unit 1 and Unit 2 TS respectively, dated February 9, 1996, implemented 10 CFR 50 Appendix J, Option B. The amendment change approved a Containment Leakage Rate Testing Program based on 10 CFR 50 Appendix J, Option B that was outside of the CTS and did not include these two specifications in that program, but retained them in CTS 4.6.1.1. Since these specifications contain specific testing requirements not contained in 10 CFR 50 Appendix J, Option B, they probably should be retained in the ITS as SRs in ITS 3.6.1 and 3.6.3. Since the STS does not contain a specific SR for equipment hatch leakage other than what may be implied by STS SR 3.6.1.1 and SR 3.6.2.1, it may be possible to provide a justification to relocate CTS 4.6.1.1.c out of the ITS. However, this change would be considered as a beyond scope of review item for this conversion. As for CTS 4.6.1.1.d, the STS does contain a SR on purge valve leakage. TSTF-52 Rev. 3 did not remove or relocate the purge valve leakage SR (STS SR 3.6.3.7). Refer to Comment Number 3.6.3-2 for justification for including this specification in the ITS. Also, refer to Comment Numbers 3.6.1-4 and 3.6.1-6. **Comment:** Revise the CTS/ITS markup to retain these two specifications. Provide the appropriate discussions and justifications for any changes made in converting to the ITS.

Response: The Company will take the action proposed in the Comment, with certain modifications.

The CTS 4.6.1.1.d markup is modified, adopting the requirement as modified and justified by DOC A.12, DOC A.13, and DOC LA.4, adopting ISTS SR 3.6.3.7, as modified and justified by JFD 10.

The CTS 4.6.1.1.c markup is retained in Chapter 5.0, and details regarding how to satisfy Surveillance Requirements after each closing of the equipment hatch are moved to the

Containment Leakage Rate Testing Program as justified by DOC LA.12. This change is acceptable because Regulatory Guide 1.163, dated September 1995, required by ITS 5.5.15, states that NEI 94-01, Revision 0, provides methods acceptable to the NRC for complying with 10 CFR Part 50, Appendix J, Option B. Section 10.2.1.3 of NEI 94-01 requires a Type B test be performed prior to the time containment integrity is required, if a containment penetration is opened. The equipment hatch is a containment penetration, so it must be tested prior to the time containment integrity is required. The company does not consider this a beyond scope change because this adopts the format and requirements of the ISTS, which are consistent with the design of the equipment hatch.

ITS Chapter 5.0 markups are modified and DOC A.26 is deleted to reflect these changes.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.3.3	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program	
SR 3.6.3.4	Perform leakage rate testing for containment purge valves with resilient seals.	Prior to entering MODE 4 from MODE 5 after containment vacuum has been broken	RAI 3.6.3-2 RAI 3.6.1-5 R1
SR 3.6.3.5	Verify each automatic power operated containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months	RAI 3.6.3-2 RAI 3.6.1-5 R1
SR 3.6.3.6	Cycle each weight or spring loaded check valve not testable during operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is < 1.2 psid and opens when the differential pressure in the direction of flow is ≥ 1.2 psid and < 5.0 psid.	18 months	RAI 3.6.3-2 RAI 3.6.1-5 R1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.4

RAI 3.6.3-2 RAI 3.6.1-5 R1

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types.

This SR must be performed prior to entering MODE 4 from MODE 5 after containment vacuum has been broken. This Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). This Frequency will ensure that each time these valves are cycled they will be leak tested.

SR 3.6.3.5

Automatic power operated containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic power operated containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.6

RAI 3.6.3-2 RAI 3.6.1-5 R1

The check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.6 verifies the operation of the check valves that are not testable during unit operation. The Frequency of 18 months is based on such factors as the inaccessibility of

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.6.3.6 (continued) these valves, the fact that the unit must be shut down to perform the tests, and the successful results of the tests on an 18 month basis during past unit operation.	RAI 3.6.3-2 RAI 3.6.1-5 R1
REFERENCES	1. UFSAR, Chapter 15. 2. Technical Requirements Manual. 3. Standard Review Plan 6.2.4. 4. UFSAR, Section 6.2.4.2.	RAI 3.6.3-14 R1

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.6.3.~~7~~ 4	Perform leakage rate testing for containment purge valves with resilient seals.	Prior to entering MODE 4 from MODE 5 after containment vacuum has been broken ~~184 days AND Within 92 days after opening the valve~~	(5) (10) (2) RAI 3.6.3-2 3.6.1-5 R1
4.6.3.1.2.a, b & c SR 3.6.3.~~8~~ 5	Verify each automatic power operated containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	~~[~~18~~]~~ months	(4) (5) (2)
4.6.3.1.2.d SR 3.6.3.~~9~~ 6	Cycle each weight or spring loaded check valve not testable during operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is ≤ ~~[~~1.2~~]~~ psid and opens when the differential pressure in the direction of flow is ≥ ~~[~~1.2~~]~~ psid and < ~~[~~5.0~~]~~ psid.	18 months	(5) (2) (2)
~~SR 3.6.3.10~~	~~Verify each [ ] inch containment purge valve is blocked to restrict the valve from opening > [50]%.~~	~~[18] months~~	(9) RAI 3.6.3-12 R1

(continued)

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(2)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.~~7~~ (4) (2)

RAI 3.6.3-2 3.6.1-4 3.6.1-5 R1

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. TSTF-52 Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. ~~Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3).~~ (2) (3)

6

~~Additionally,~~ this SR must be performed ~~within 92 days after opening the valve~~ prior to entering MODE 4 from MODE 5 after containment vacuum has been broken. This ~~The 92 day~~ Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). ~~Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.~~ This Frequency will ensure that each time these valves are cycled they will be leak tested (2) (2)

SR 3.6.3.~~8~~ (5) (2)

Automatic power operated containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~plant~~ unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. (3) (1) (3)

(continued)

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(2)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.9 6

RAI 3.6.3-2 3.6.1-5

In subatmospheric containments, the check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.9 (6) verifies the operation of the check valves that are not testable during unit operation. The Frequency of 18 months is based on such factors as the inaccessibility of these valves, the fact that the unit must be shut down to perform the tests, and the successful results of the tests on an 18 month basis during past unit operation. (2)

SR 3.6.3.10

Reviewer's Note: This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.

Verifying that each [42] inch containment purge valve is blocked to restrict opening to ≤ [50]% is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage. (2)

SR 3.6.3.11

This SR ensures that the combined leakage rate of all shield building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the (2)

(continued)

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A.1

ITS 3.6.3

ITS

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained. <See ITS 3.6.1>

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated: <See ITS 3.6.1>

SR 3.6.3.1 / SR 3.6.3.2 — a. INSERT Proposed SR 3.6.3.1 Note — At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves, secured in their positions, except for valves that are locked, sealed, or otherwise secured, or are open under administrative control as permitted by Specification 3.6.3.1. L.5 L.6 RAI 3.6.3-9 R1

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3. <See ITS 3.6.1>

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals, with gas at P_a, greater than or equal to 44.1 psig. Results shall be evaluated against the criteria of Specification 3.6.1.2.b as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. <See ITS 5.0>

SR 3.6.3.4 — d. ~~Each time containment integrity is established~~ Prior to entering MODE 4 from MODE 5 after vacuum has been broken by pressure testing the ~~butterfly~~ isolation valves with resilient seals in the containment purge lines and the ~~containment vacuum ejector line~~. A.12 LA.4 A.13 RAI 3.6.3-2 3.6.1-5 R1

SR 3.6.3.1 — * Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are not locked sealed or otherwise sealed in the closed position. L.7 RAI 3.6.3-9

SR 3.6.3.2 — ~~These penetrations shall be verified closed during each COLD SHUTDOWN except that such surveillance need not be performed more often than once per 92 days.~~ INSERT Proposed SR 3.6.3.2 Note L.5 R1

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ITS 3.6.3

ITS

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

<See ITS 3.6.1>

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated: <See ITS 3.6.1>

SR 3.6.3.1
SR 3.6.3.2

INSERT Proposed SR 3.6.3.1 Note

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves, secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1. locked, sealed, or otherwise secured, or

L.5
L.6 RAI 3.6.3-9 R1

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3. <See ITS 3.6.1>

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals, with gas at P_a, greater than or equal to 44.1 psig. Results shall be evaluated against the criteria of Specification 3.6.1.2.b as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. <See ITS 5.0>

SR 3.6.3.4

d. Each ~~time~~ containment integrity is established after vacuum has been broken by pressure testing the butterfly isolation valves with resilient seals in the containment purge lines and the containment ~~vacuum ejector~~ line. Prior to entering MODE 4 from MODE 5

A.12
LA.4
A.13
RAI 3.6.3-2 3.6.1-5 R1

L.7 RAI 3.6.3-9 R1

SR 3.6.3.1
SR 3.6.3.2

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are not locked sealed or otherwise sealed in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such surveillance need not be performed more often than once per 92 days.

Insert proposed SR 3.6.3.2 Note

L.5