Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS 3.6.9 Bases

Description

In the North Anna ITS, the term, "other unit" is used to refer to the other station unit. The ITS 3.6.9 Bases used the term "opposite unit." This is changed to "other unit" for consistency with the remainder of the North Anna ITS. This affects the ITS Bases and ISTS Bases markup.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and UFSAR, Chapter 3, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor is returned to containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. The two systems are shared with the other unit. R13
Each system consists of controls located in the recombiner vault, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture to greater than or equal to 1100°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Emergency Diesel Generator bus, is capable of being powered from any Emergency Diesel Generator bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 (a) Hydrogen Recombiners ~~(Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~ ~~(if permanently installed)~~ (1)

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, ~~"Containment Atmosphere Cleanup"~~ (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) ~~or steam line break (SLB)~~. The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor ~~remains in~~ is returned to containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA). (6) (2)

Two 100% capacity independent hydrogen recombiner systems are provided. The two systems are shared with the other unit. Each system consists of controls located in the ~~control room~~ recombiner vault, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen air mixture ~~above 1150~~ to greater than or equal to 1100°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen concentration in containment below the ~~4.1~~ 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate ~~Engineered Safety Features~~ Emergency Diesel Generator bus, ~~and is provided with a separate power panel and control panel.~~ is capable of being powered from any Emergency Diesel Generator bus. (7) 1R13 (2) (2) (2)

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of ~~4.1~~ 4.0 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. (2)

(continued)

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Specifications Affected: CTS Markup for ITS 3.6.6

Description

Page 1 of the Unit 1 and Unit 2 CTS markup of ITS 3.6.6 stated that Surveillance 4.6.2.1.a.2 was discussed in ITS 3.5.4. This Surveillance is not discussed in ITS 3.5.4. The reference to ITS 3.5.4 is removed and DOC A.2 is added to address the elimination of Surveillance 4.6.2.1.a.2.

The ITS and Bases are unaffected.

ITS

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6 3.6.2.1 Two ~~independent~~ containment quench spray subsystems shall be OPERABLE. LA.1

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Action A Action B With one containment quench spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment quench spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

SR 3.6.6.1 1. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position. A.2 | R13

2. ~~Verifying the temperature of the borated water in the refueling water storage tank is within the limits shown on Figure 3.6-1.~~ LA.5

SR 3.6.6.2 b. Verifying ~~that on recirculation flow~~ each pump develops a discharge ~~pressure of~~ head ≥ ~~123 psig~~ when tested pursuant to Specification 4.0.5. the required developed head at the flow test point LA.2

c. At least once per 18 months during shutdown, by:

SR 3.6.6.3 1. Verifying that each automatic valve in the flow path That is not locked, sealed, or otherwise secured in position actuates to its correct position on ~~a containment high-high~~ an actual or simulated actuation signal. L.1 LA.3

SR 3.6.6.4 2. Verifying that each spray pump starts automatically on ~~a containment high-high~~ an actual or simulated actuation signal. L.2 LA.3

A.1

ITS 3.6.6

05-16-94

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

ITS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

.6.6 3.6.2.1 Two ~~independent~~ containment quench spray subsystems shall be OPERABLE. LA.1

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

ction A With one containment quench spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and

tion B in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment quench spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by:

SR 3.6.6.1 1. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

~~2. Verifying the temperature of the borated water in the refueling water storage tank is within the limits shown on Figure 3.6-1.~~ A.2 | R13

LA.5

SR 3.6.6.2 b. Verifying ~~that on recirculation flow,~~ each pump develops a discharge ~~pressure of~~ head greater than or equal to ~~123 psig~~ the required developed head at the flow test point when tested pursuant to Specification 4.0.5. LA.2

c. At least once per 18 months during shutdown, by: LA.3

SR 3.6.6.3 1. Verifying that each automatic valve in the flow path that is not locked, sealed, or otherwise secured in position actuated to its correct position on ~~a Containment Pressure – high-high~~ an actual or simulated actuation signal.

SR 3.6.6.4 2. Verifying that each spray pump starts automatically on ~~a Containment Pressure – high-high~~ signal. L.2

SR 3.6.6.5 d. At least once per 10 years by ~~performing an air or smoke flow test through each spray header and~~ verifying each spray nozzle is unobstructed. LA.3 | LA.4

L.1

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ADMINISTRATIVE CHANGES

A.1 In the conversion of the North Anna Current Technical Specifications (CTS) to the plant specific Improved Technical Specifications (ITS), certain changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) are made to obtain consistency with NUREG-1431, Rev. 1, "Standard Technical Specifications-Westinghouse Plants" (ISTS).

These changes are designated as administrative changes and are acceptable because they do not result in technical changes to the CTS.

A.2 CTS 4.6.2.1.a.2 states that the temperature of the borated water in the refueling water storage tank must be verified to be within the limits shown on Figure 3.6-1 every 31 days. Figure 3.6-1 states that the RWST temperature must be ≤ 50°F. CTS LCO 3.6.1.4 and CTS 4.6.1.4 require that the containment internal air partial pressure be in compliance with Figure 3.6-1 every 12 hours. CTS 3.5.5.c and 4.5.5.b require verification that the RWST temperature is between 40° and 50° every 7 days. ITS SR 3.5.4.1 requires verification of RWST temperature every 24 hours. This changes the CTS by eliminating the verification of RWST temperature every 7 days from the quench spray requirements. R13

This change is acceptable because the technical requirements have not changed. The requirements to verify RWST temperature in CTS 3.6.1.4, CTS 3.5.5, ITS SR 3.6.4.1, and ITS SR 3.5.4.1 are performed more frequently than the requirement in CTS 4.6.2.1.a.2. Therefore, the elimination of this Surveillance has no effect on plant operation. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS LCO 3.6.2.1 states that two independent containment quench spray subsystems shall be OPERABLE. ITS 3.6.6 also requires two quench spray

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS 3.7.5 Bases

Description

The Bases to ITS 3.7.5, Required Action C.2, states that the Completion Time is 12 hours. The correct Completion Time, as stated in ITS 3.7.5 Required Action C.2, is 18 hours. The Bases are corrected. This affects the ITS 3.7.5 Bases. The ISTS Bases markup is correct.

BASES

ACTIONS

B.1 (continued)

capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

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

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

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours. R13

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, when the steam generator is relied upon for heat removal, with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the unit should not be

(continued)

Specifications Affected: ITS 3.7.5 CTS markup and DOCs

Description

The CTS markup for ITS 3.7.5, Required Actions C.1 and C.2 is incorrect. The CTS requires being in MODE 4 within 6 hours. The ITS requires being in MODE 3 in 6 hours and MODE 4 in 18 hours. This was incorrectly characterized as a more restrictive change. The CTS markup is corrected, DOC M.1 is revised, and DOC L.10 is added to address the change to the CTS.

The ITS and Bases are unaffected.

ITS

PLANT SYSTEMS

A.1

AUXILIARY FEEDWATER SYSTEM (AFW)

LIMITING CONDITION FOR OPERATION

LCO 3.7.5 3.7.1.2 At least three ~~independent~~ steam generator auxiliary feedwater ~~pumps and associated flow paths~~ trains shall be OPERABLE ~~with:~~ A.3 LA.1

~~a. Two motor driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and~~

~~b. One steam turbine driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.~~

add proposed LCO Note M.1

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

MODE 4 when steam generator is relied upon for heat removal M.1

ACTION:

add proposed Action A L.8

Action A
Action B
Action C

a. With one auxiliary feedwater ~~pump~~ inoperable, restore the required auxiliary feedwater ~~pumps~~ to an OPERABLE status within 72 hours and 10 days from discovery to meet the LCO or ~~be in at least HOT SHUTDOWN within the following 6 hours.~~ be in MODE 3 in 6 hours & MODE 4 in 18 hours A.3 L.10 R13

A.1 add first proposed Condition C Trains A.3 in Modes 1, 2, 3

b. With two auxiliary feedwater ~~pumps~~ inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within ~~the following 6~~ 18 hours. L.9

add proposed Note to Action D L.6

Action D c. With three auxiliary ~~feedwater pumps~~ trains inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater ~~pump~~ train to OPERABLE status as soon as possible. A.3

Action E add proposed Action E M.1

SURVEILLANCE REQUIREMENTS

A.1

~~4.7.1.2 In addition to the requirements of Specification 4.0.5, each auxiliary feedwater pump shall be demonstrated OPERABLE:~~

a. At least once per 31 days by: A.2

SR 3.7.5.1 1. Verifying that each valve (manual, power operated or automatic) in the flow path and both steam supply flow paths to the steam turbine driven pump that is not locked, sealed, or otherwise secured in position, is in its correct position. L.2

add proposed Note to SR 3.7.5.2 L.4

SR 3.7.5.2 b. ~~At least once per 92 days on a STAGGERED TEST BASIS by:~~ Test in accordance with the IST program

1. Verifying that each pump develops adequate discharge pressure and flow. ~~The acceptance criterion shall be consistent with Specification 4.0.5.~~ The provisions of Specification 4.0.4 are not applicable to steam turbine driven pump testing. L.2

Note in SR 3.7.5.2

ITS

PLANT SYSTEMS (A.1)

AUXILIARY FEEDWATER SYSTEM (AFW)

LIMITING CONDITION FOR OPERATION (A.3)

LCO 3.7.5 3.7.1.2 At least three ~~independent~~ steam generator auxiliary feedwater ~~pumps and associated flow paths~~ trains shall be OPERABLE ~~with:~~ (LA.1)

a. ~~Two motor driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and~~

b. ~~One steam turbine driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.~~

add proposed LCO Note (M.1)

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

ACTION:

Action A — add proposed Action A (L.8) (L.3)

Action A
Action B
Action C

a. With one auxiliary feedwater ~~pump~~ train (A.3) inoperable restore the required auxiliary feedwater ~~pumps~~ to an OPERABLE status within 72 hours in MODES 1, 2, or 3 and 10 days from discovery to met the LCO, (L.8) or be in ~~at least HOT SHUTDOWN within the following 6 hours.~~ MODE 3 in 6 hours & MODE 4 within 18 hours (L.10) RB

(A.1) add first proposed Condition C

b. With two auxiliary feedwater ~~pumps~~ trains (A.3) inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within ~~the following 6~~ 18 hours. (L.9)

add proposed Note to Action D

Action D c. With three auxiliary feedwater ~~pumps~~ trains inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater ~~pump~~ train (A.3) to OPERABLE status as soon as possible. (L.6)

Action E — add proposed Action E (M.1)

SURVEILLANCE REQUIREMENTS

~~4.7.1.2 In addition to the requirements of Specification 4.0.5, each auxiliary feedwater pump shall be demonstrated OPERABLE:~~ (A.1)

SR 3.7.5.1 a. At least once per 31 days by:

1. Verifying that each valve (manual, power operated or automatic) in the flow path and both steam supply flow paths to the steam driven pump (A.2) that is not locked, sealed, or otherwise secured in position, is in its correct position.

add proposed Note to SR 3.7.5.2 (L.2)

SR 3.7.5.2 b. ~~At least once per 92 days on a STAGGERED TEST BASIS by:~~ Test in accordance with IST program (L.4)

1. Verifying that each pump develops adequate discharge pressure ~~and flow.~~ ~~The acceptance criterion shall be consistent with Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable to steam turbine driven pump testing.~~ (L.2)

Note in SR 3.7.5.2

page 1 of 2

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removal for the system. To support this change in the Applicability, the following additional requirements are added to the CTS:

- A note is added to the LCO that requires an AFW train, supported by a motor driven pump, to be operable in MODE 4; | R13
- A new ACTION E is added which requires an immediate action to restore a required inoperable AFW train to OPERABLE status when the SG is required in MODE 4; and
- The addition of Notes to ITS SRs 3.7.5.3 and 3.7.5.4 which state the requirements are not applicable in MODE 4 when a steam generator is relied upon for heat removal.

These changes are acceptable because they ensure the necessary support systems are available when a steam generator is being relied upon for heat removal in Mode 4. The CTS do not have specific requirements for an AFW train to be OPERABLE in MODE 4 when a steam generator is relied upon for heat removal. The definition of OPERABILITY is contained in Section 1.0 of the ITS and requires the applicable systems to be OPERABLE to support the required function. In this case, the AFW system is required to support the SG. These changes clarify this requirement. One AFW train, supplied by a motor driven pump, will provide sufficient water to the SG to remove decay heat in MODE 4. If the AFW train is inoperable, ITS ACTION E requires the initiation of action to restore the AFW train to OPERABLE status immediately. This is acceptable because without the SG it may not be possible to cool down the unit and exit the MODE of applicability. Additionally, during MODE 4, the OPERABLE AFW train does not need to be capable of being placed in service automatically. Manual operation of the system is acceptable, because the heat removal requirements are less in MODE 4. Thus, there would be sufficient time for the operators to diagnose and respond to an RCS temperature excursion. These changes are designated as more restrictive because they place additional requirements on plant operations in MODE 4 that are not required by the CTS.

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description)* CTS LCO 3.7.1.2 requires three independent AFW pumps and associated flow paths to be OPERABLE. This includes the motor driven AFW pumps powered from separate emergency buses, and the steam turbine driven AFW pump capable of being powered from an OPERABLE steam supply system. ITS LCO 3.7.5 will require "Three AFW trains to be OPERABLE"; it does not include design details or define the components

AFW trains inoperable, be in MODE 3 in 6 hours and MODE 4 in 18 hours. This changes the CTS by allowing 18 hours instead of 12 hours to be in MODE 4.

This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. The allowance to place the plant in MODE 4 in 18 hours allows the unit to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. The time frame of 18 hours to require the plant to move from 100 % power to MODE 4 is consistent with other CTS and ITS requirements when the heat removal capability of unit is degraded. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS.

L.10 *(Category 3 – Relaxation of Completion Time)* CTS 3.7.1.2 ACTION a. states that with one AFW pump inoperable, restore the pump to OPERABLE status within 72 hours or be in HOT SHUTDOWN (i.e., MODE 4) within 6 hours. ITS ACTION C states that if an inoperable AFW train is not restored, be in MODE 3 in 6 hours and MODE 4 in 18 hours. This changes the CTS by allowing 6 hours to be in MODE 3 instead of MODE 4 and allowing 18 hours to be in MODE 4.

This change is acceptable because the Completion Time is consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, a reasonable time for repairs or replacement, and the low probability of a DBA occurring during the allowed Completion Time. The allowance to place the plant in MODE 3 in 6 hours and MODE 4 in 18 hours allows the unit to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. The time frame of 18 hours to require the plant to move from 100 % power to MODE 4 is consistent with other ITS requirements when the heat removal capability of the unit is degraded. This change is designated as less restrictive because additional time is allowed to restore parameters to within the LCO limits than was allowed in the CTS. R13

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS 3.7.8 Bases

Description

The ITS 3.7.8 LCO Bases is revised to state an additional Service Water configuration that supports an OPERABLE Service Water loop. A Service Water loop is OPERABLE if there is one OPERABLE Service Water pump instead of two, provided both Service Water pumps in the other Service Water train are OPERABLE, and Service Water flow is throttled to the Component Cooler heat exchangers. This is consistent with the plant design, the CTS, and the Bases for ACTION A.1. This ITS Bases and ISTS Bases markup are affected by this change.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The SW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two SW loops are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

A SW loop is considered OPERABLE during MODES 1, 2, 3, and 4 when:

a. Either R10 R13

a.1 Two SW pumps are OPERABLE in an OPERABLE flow path; or

a.2 One SW pump is OPERABLE in an OPERABLE flow path provided two SW pumps are OPERABLE in the other loop and SW flow to the CC heat exchangers is throttled; and R13

b. Three spray arrays are OPERABLE in an OPERABLE flow path; and R10

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

For two SW loops to be considered OPERABLE during MODES 1, 2, 3, and 4, the following conditions must also be met in order to provide protection for a single active failure of the actuation circuitry: R10

a. With one SW pump operating on each SW loop, the operating pumps have opposite train designations; and

b. With one of the four spray arrays on each SW loop inoperable, the inoperable spray arrays have opposite train designations.

A required valve directing flow to a spray array, bypass line, or other component is considered OPERABLE if it is capable of automatically moving to its safety position or if it is administratively placed in its safety position.

INSERT 1

a. Either

a.1. Two SW pumps are OPERABLE in an OPERABLE flow path; or

a.2 One SW pump is OPERABLE in an OPERABLE flow path provided two SW pumps are OPERABLE in the other loop and SW flow to the CC heat exchangers is throttled; and

R13

b. Three spray arrays are OPERABLE in an OPERABLE flow path; and

R10

INSERT 2

A.1

If one SW pump is inoperable, the flow resistance of the system must be adjusted within 72 hours by throttling component cooling water heat exchanger flows to ensure that design flows to the RS System heat exchangers are achieved following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. In this configuration, a single failure disabling a SW pump would not result in loss of the SW System function.

B.1 and B.2

If one or more SW System loops are inoperable due to only two SW pumps being OPERABLE, the flow resistance of the system must be adjusted within one hour to ensure that design flows to the RS System heat exchangers are achieved if no additional failures occur following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. Two SW pumps aligned to one loop or one SW pump aligned to each loop is capable of performing the safety function if CC heat exchanger flow is properly throttled. However, overall reliability is reduced because a single failure disabling a SW pump could result in loss of the SW System function. The one hour time reflects the need to minimize the time that two pumps are inoperable and CC heat exchanger flow is not properly throttled, but is a reasonable time based on the low probability of a DBA occurring during this time period. Restoring one SW pump to OPERABLE status within 72 hours together with the throttling ensures that design flows to the RS System heat exchangers are achieved following an accident. The required resistance is obtained by throttling SW flow through the CC heat exchangers. In this configuration, a single failure disabling a SW pump would not result in loss of the SW System function.

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS 3.7.10, 3.7.11, 3.7.13, and Chapter 5.0 CTS markup

Description

The ITS submittal included proposed CTS changes for the ventilation system requirements which were submitted to the NRC in a separate license amendment. This license amendment was approved as Amendment 228 (Unit 1) and 209 (Unit 2) on December 1, 2001. There were no differences between the proposed CTS pages used in the ITS submittal and the approved CTS pages. The proposed CTS pages are replaced with the approved CTS pages.

The ITS and ITS Bases are unaffected.

A.1

ITS

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

SR 3.7.10.1

a. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ by initiating, ~~from the control room,~~ flow through ~~the HEPA filters and charcoal adsorbers~~ and verifying that the system operates for at least 10 hours with the heaters on. L.1 LA.1 A.2

Insert proposed SR 3.7.10.2

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10% (except as shown in Specifications 4.7.7.1e. and f.).

2. Verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. Within 31 days of completing 720 hours of charcoal adsorber operation, verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

See ITS 5.0

SR 3.7.10.3
SR 3.7.10.4

d. At least once per 18 months by:

1. Verifying that the pressure drop across the demister filter, HEPA filter and charcoal adsorber is < 4 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

A.1

each LCO 3.7.10.a MCR/ESGR EVS train actuates

M.5

ITS

PLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.7.10.3 2. Verifying that the ~~normal~~ air supply ~~and exhaust are~~ automatically shutdown on a Safety Injection Actuation Test Signal. on an actual or simulated actuation LA.2 L.2

every 18 months on a STAGGERED TEST BASIS

SR 3.7.10.4 3. Verifying that ~~the system~~ each required train maintains the control room at a positive pressure of ≥ 0.04 inch W. G. relative to the ~~outside atmosphere~~ adjacent areas at a system flow rate of 1000 cfm ± 10%. L.3 M.7 M.6

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove ≥ 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 102 bottles of air (shared with Unit 2) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of ≥ 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

See ITS 3.7.13

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is ≤ 120°F.

See ITS 3.7.11

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PLANT SYSTEMS

ITS

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

L.1 LA.1 A.2

a. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ by initiating, ~~from the control room, flow through the HEPA filters and charcoal adsorbers~~ and verifying that the system operates for at least 10 hours with the heaters on.

INSERT PROPOSED SR 3.7.10.2

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10% (except as shown in Specifications 4.7.7.1e. and f.).

2. Verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

See ITS 5.0

c. Within 31 days of completing 720 hours of charcoal adsorber operation, verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

SR 3.7.10.3
SR 3.7.10.4

d. At least once per 18 months by:

1. Verifying that the pressure drop across the demister filter, HEPA filter and charcoal adsorber assembly is < 4 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

A.1

each LCO 3.7.10.a MCR/ESGR EVS train actuates

M.5

PLANT SYSTEMS

ITS

SURVEILLANCE REQUIREMENTS (Continued)

LA.2

SR 3.7.10.3 every 18 months on a STAGGERED TEST BASIS

2. Verifying that ~~the normal air supply and exhaust are~~ automatically ~~shutdown on a~~ Safety Injection Actuation Test Signal. on an actual or simulated actuation

L.2

L.3

SR 3.7.10.4

3. Verifying that ~~the system~~ maintains the control room at a positive pressure of greater than or equal to 0.04 inch W. G. relative to the ~~outside atmosphere~~ adjacent areas at a system flow rate of 1000 cfm ± 10%.

each required train

M.7 M.6

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 102 bottles of air (shared with Unit 1) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of greater than or equal to 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

See ITS 3.7.13

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F.

See ITS 3.7.11

ITS

PLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the normal air supply and exhaust are automatically shutdown on a Safety Injection Actuation Test Signal.

3. Verifying that the system maintains the control room at a positive pressure of ≥ 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm ± 10%.

See ITS 3.7.10

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove ≥ 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 102 bottles of air (shared with Unit 2) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of ≥ 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

See ITS 3.7.13

SR 3.7.11.1 4.7.7.3 Each ~~control room~~ air-conditioning system ~~shall be demonstrated OPERABLE at least once per 12 hours by~~ verifying that the control ~~room~~ air temperature is ≤ 120°F.

INSERT PROPOSED SR 3.7.11.1 M.2

A.1

ITS

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the normal air supply and exhaust are automatically shutdown on a Safety Injection Actuation Test Signal.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm ± 10%.

See ITS 3.7.10

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 102 bottles of air (shared with Unit 1) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of greater than or equal to 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

See ITS 3.7.13

4.7.7.3 Each ~~control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F.~~

SR 3.7.11.1 INSERT PROPOSED SR 3.7.11.1 M.2

(A.1)

ITS 3.7.13

ITS

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

See ITS 3.7.10

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10% (except as shown in Specifications 4.7.7.1e. and f.).

See ITS 5.0

2. Verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. Within 31 days of completing 720 hours of charcoal adsorber operation, verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

SR 3.7.13.3
SR 3.7.13.4

d. At least once per 18 months by:

1. Verifying that the pressure drop across the demister filter, HEPA filter and charcoal adsorber is < 4 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

A.1

ITS 3.7.13

each required MCR/ESGR bottled air system train actuates

M.5

PLANT SYSTEM

ITS

SURVEILLANCE REQUIREMENTS (Continued)

On an actual or simulated actuation

L.2

SR 3.7.13.3

2. Verifying that ~~the normal air supply and exhaust are~~ automatically ~~shutdown on a Safety Injection Actuation Test~~ Signal.

LA.2

3. Verifying that the system maintains the control room at a positive pressure of ≥ 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm ± 10%.

See ITS 3.7.10

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove ≥ 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

M.6

SR 3.7.13.2

SR 3.7.13.1

a. At least once per 31 days by verifying that ~~the system contains a minimum of 102 bottles of air (shared with Unit 2)~~ each pressurized to at least 2300 psig.

LA.3

two required trains

SR 3.7.13.4

b. At least once per 18 months by verifying that ~~the system~~ will supply at least 340 cfm of air to maintain the control room at a positive pressure of ≥ 0.05 inch W.G. relative to the ~~outside atmosphere~~ for at least 60 minutes.

M.7

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is ≤ 120°F.

See ITS 3.7.11

On a STAGGERED TEST BASIS

L.3

adjacent areas

M.8

each required MCR/ESGR bottled air bank manual valve not locked, sealed, or otherwise secured, and required to be open during accident conditions is open

M.6

A.1

ITS 3.7.13

ITS

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

See ITS 3.7.10

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10% (except as shown in Specifications 4.7.7.1e. and f.).

2. Verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

See ITS 5.0

c. Within 31 days of completing 720 hours of charcoal adsorber operation, verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

SR 3.7.13.3
SR 3.7.13.4

d. At least once per 18 months by:

1. Verifying that the pressure drop across the demister filter, HEPA filter and charcoal adsorber assembly is < 4 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%.

See ITS 5.0

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(A.1) ITS 3.7.13

each required MCR/ESGR bottled air system train actuates (M.5)

PLANT SYSTEMS

ITS

SURVEILLANCE REQUIREMENTS (Continued)

on an actual or simulated actuation (L.2)

SR 3.7.13.3 2. Verifying that ~~the normal air supply and exhaust are~~ automatically shutdown on a ~~Safety Injection Actuation Test~~ Signal. (LA.2)

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm ± 10%. See ITS 3.7.10

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%. See ITS 5.0

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

SR 3.7.13.2 a. At least once per 31 days by verifying that ~~the system contains a minimum of~~ (M.6)

SR 3.7.13.1 ~~102 bottles of air~~ ~~(shared with Unit 1)~~ each pressurized to at least 2300 psig. | (LA.3)

two required trains

SR 3.7.13.4 b. At least once per 18 months by verifying that ~~the system~~ will supply at least 340 cfm of air to maintain the control room at a positive pressure of greater than or equal to 0.05 inch W.G. relative to the ~~outside atmosphere~~ for at least 60 minutes. (M.7)

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F. See ITS 3.7.11

on a STAGGERED TEST BASIS (L.3)

adjacent areas (M.8)

each required MCR/ESGR bottled air bank manual valve not locked, sealed, or otherwise secured, and required to be open during accident conditions is open (M.6)

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(A.1) ITS 5.0

ITS

INSERT →

A.5
A.23

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE: See ITS 3.7.10

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per ~~18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system~~ by: LA.5

5.5.10.a 5.5.10.b 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10% ~~(except as shown in Specifications 4.7.7.1e and f.)~~. LA.5

5.5.10.c 2. Verifying, ~~within 31 days after removal,~~ that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.10.a 5.5.10.b 3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

5.5.10.c c. ~~Within 31 days of completing 720 hours of charcoal adsorber operation,~~ verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%. LA.5

d. ~~At least once per 18 months by:~~ LA.5

5.5.10.d 1. Verifying that the pressure drop across the demister filter, HEPA filter and charcoal adsorber is < 4 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%. RAI 5.0-06 R4

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ITS INSERT →

A.5

A.23

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE: See ITS 3.7.10

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by: LA.5

5.5.10.a 1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000

5.5.10.b cfm ± 10% (except as shown in Specifications 4.7.7.1e. and f.). LA.5

5.5.10.c 2. Verifying, within 31 days after removal, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.

5.5.10.a 5.5.10.b 3. Verifying a system flow rate of 1000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. Within 31 days of completing 720 hours of charcoal adsorber operation, verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 2.5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%. LA.5 5.5.10.c

d. At least once per 18 months by: LA.5

5.5.10.d 1. Verifying that the pressure drop across the demister filter, HEPA filter and charcoal adsorber assembly is < 4 inches Water Gauge while operating the filter train at a flow rate of 1000 cfm ± 10%. RAI 5.0-06 R4

PLANT SYSTEMS

ITS

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the normal air supply and exhaust are automatically shutdown on a Safety Injection Actuation Test Signal.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 0.04 inch W. G. relative to the outside atmosphere at a system flow rate of 1000 cfm ± 10%.

See ITS 3.7.10

5.5.10.a e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%. LA.5

5.5.10.b f. After each complete or partial replacement of a charcoal adsorber bank by verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%. LA.5

4.7.7.2 The bottled air pressurization system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the system contains a minimum of 102 bottles of air (shared with Unit 1) each pressurized to at least 2300 psig.

b. At least once per 18 months by verifying that the system will supply at least 340 cfm of air to maintain the control room at a positive pressure of greater than or equal to 0.05 inch W.G. relative to the outside atmosphere for at least 60 minutes.

See ITS 3.7.13

4.7.7.3 Each control room air-conditioning system shall be demonstrated OPERABLE at least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F.

See ITS 3.7.11

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS 3.7.12 Bases

Description

The Bases of ITS 3.7.12 are revised. The ECCS PREACS trains consist of two subsystems; the Safeguards Area Ventilation subsystem and the Auxiliary Building Central Exhaust subsystem. The description of the ECCS PREACS is revised and reorganized to discuss each subsystem separately. The Auxiliary Building Central Exhaust subsystem is shared between Units 1 and 2 and both trains share common ductwork. As a result, any opening of the ductwork could be construed as rendering both trains of the Auxiliary Building Central Exhaust subsystem inoperable for both units. This would prevent any maintenance, repair or testing on the Auxiliary Building Central Exhaust subsystem unless both units are shutdown. This is inconsistent with current operating procedures. The Auxiliary Building Central Exhaust subsystem does not appear in the CTS. The accident analyses assume that filtration by the ECCS PREACS does not begin for 60 minutes following an accident. Plant procedures allow the Auxiliary Building Central Exhaust subsystem to be considered OPERABLE during maintenance, testing and repair provided the subsystem can be restored to service following an accident before the ECCS PREACS is needed. That interpretation is incorporated into the ITS Bases. This affects the ITS Bases, the ISTS Bases markup, and adds ITS Bases JFD 14.

B 3.7 PLANT SYSTEMS

B 3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA). The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature in the ECCS pump room areas.

The ECCS PREACS consists of two subsystems, the Safeguards Area Ventilation subsystem and the Auxiliary Building Central Exhaust subsystem. There are two redundant trains in the Safeguards Area Ventilation subsystem. Each train of the Safeguards Area Ventilation subsystem consists of one Safeguards Area exhaust fan, prefilter, and high efficiency particulate air (HEPA) filter and charcoal adsorber assembly for removal of gaseous activity (principally iodines) (shared with the other unit), and controls for the Safeguards Area exhaust filter and bypass dampers. Ductwork, valves or dampers, and instrumentation also form part of the subsystem. The subsystem automatically initiates filtered ventilation of the safeguards pump room following receipt of a Containment Hi-Hi signal from the affected unit. R13

The Auxiliary Building Central exhaust subsystem consists of the following: three redundant central area exhaust fans (shared with other unit), two redundant filter banks consisting of HEPA filter and charcoal adsorber assembly for removal of gaseous activity (principally iodines) (shared with the other unit), and two redundant trains of controls for the Auxiliary Building Central exhaust subsystem filter and bypass dampers (shared with the other unit). Ductwork, valves or dampers, and instrumentation also form part of the subsystem. The subsystem initiates filtered ventilation of the charging pump cubicles following manual actuation.

The Auxiliary Building filter banks are shared by the Safeguards Area Ventilation subsystem and the Auxiliary Building Central Exhaust subsystem. Either Auxiliary Building filter bank may be aligned to either ECCS PREACS train. These filter banks are also used by the Auxiliary

(continued)

BASES

BACKGROUND (continued)

Building General area exhaust, fuel building exhaust, decontamination building exhaust, and containment purge exhaust. R13

One Safeguards Area exhaust fan is normally operating and dampers are aligned to bypass the HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the Safeguards Area room are diverted through the filter banks. Two Auxiliary Building Central Exhaust fans are normally operating. Air discharges from the Auxiliary Building Central Exhaust area are manually diverted through the filter banks. Required Safeguards Area and Auxiliary Building Central Exhaust area fans are manually actuated if they are not already operating. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS PREACS is discussed in the UFSAR, Section 9.4 (Ref. 1) and it may be used for normal, as well as post accident, atmospheric cleanup functions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level during normal operations, generally consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 3). The heaters are not required for post-accident conditions. R13

APPLICABLE SAFETY ANALYSES

The design basis of the ECCS PREACS is established by the large break LOCA. The system evaluation assumes ECCS leakage outside containment, such as safety injection pump leakage, during the recirculation mode. In such a case, the system limits radioactive release to within the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4), and NUREG-0800, Section 6.4 (Ref. 5). The analysis of the effects and consequences of a large break LOCA is presented in Reference 2. The ECCS PREACS also may actuate following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. The analyses assume the filtration by the ECCS PREACS does not begin for 60 minutes following an accident. R13

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

BASES

LCO

Two redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available. Total system failure could result in exceeding the control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 4), and NUREG-0800, Section 6.4 (Ref. 5).

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration are OPERABLE in both trains.

An ECCS PREACS train is considered OPERABLE when its associated:

a. Safeguards Area exhaust fan is OPERABLE;

b. One Auxiliary Building HEPA filter and charcoal adsorber assembly (shared with the other unit) is OPERABLE; R13

c. One Auxiliary Building Central exhaust system fan (shared with other unit) is OPERABLE; R13

d. Controls for the Auxiliary Building Central exhaust system filter and bypass dampers (shared with the other unit) are OPERABLE; R13

e. HEPA filter and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

f. Ductwork, valves, and dampers are OPERABLE.

The Auxiliary Building Central Exhaust subsystem may be removed from service (e.g., tag out fans, open ductwork, etc.), in order to perform required testing and maintenance. The Auxiliary Building Central Exhaust subsystem is OPERABLE in this condition if it can be restored to service and perform its function within 60 minutes following an accident. R13

In addition, the required Safeguards Area and charging pump cubicle boundaries for charging pumps not isolated from the Reactor Coolant System must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, except for those openings which are left open by design, including charging pump ladder wells.

(continued)

BASES

ACTIONS

B.1 (continued)

protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump room boundary.

C.1 and C.2

If the ECCS PREACS train(s) or ECCS pump room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. R13

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the charcoal and HEPA filters from humidity in the ambient air. The system must be operated ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that Safeguards Area exhaust flow and Auxiliary Building Central Exhaust subsystem flow, when actuated from the control room, diverts flow through the Auxiliary Building HEPA filter and charcoal adsorber assembly for the operating train. Exhaust flow is diverted R13

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.2 (continued)

manually through the filters in case of a DBA requiring their use. The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.12.3

This SR verifies that the required ECCS PREACS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.4

This SR verifies that Safeguards Area exhaust flow for the operating Safeguards Area fan is diverted through the filters on an actual or simulated actuation signal. The 18 month Frequency is consistent with that specified in Reference 3.

SR 3.7.12.5

This SR verifies the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested in a qualitative manner to verify proper functioning of each train of the ECCS PREACS. During the post accident mode of operation, the ECCS PREACS is designed to maintain a slight negative pressure in the ECCS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. A single train of ECCS PREACS is designed to maintain a negative pressure relative to adjacent areas. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).

This test is conducted with the tests for filter penetration; thus, an 18 month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 3.

REFERENCES

1. UFSAR, Section 9.4. R13

B 3.7 PLANT SYSTEMS

B 3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

BASES

BACKGROUND

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA). The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature ~~and humidity~~ in the ECCS pump room area~~, and the lower reaches of the auxiliary building~~. (5) (2)

Insert

~~The ECCS PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the pump room following receipt of a safety injection (SI) signal.~~ (14) R13

~~The ECCS PREACS is a standby system, aligned to bypass the system HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned, and fans are started to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the ECCS pump room isolate, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.~~

The ECCS PREACS ~~is~~ discussed in the (a) FSAR, Sections ~~[6.5.1]~~, ~~[9.4.5]~~, ~~and [15.6.5]~~ (Refs. 1~~, 2, and 3, respectively~~) (2) (1)
(and) ~~since~~ it may be used for normal, as well as post accident, atmospheric cleanup functions. The primary purpose of the

(continued)

R13

INSERT

The ECCS PREACS consists of two subsystems, the Safeguards Area Ventilation subsystem and the Auxiliary Building Central Exhaust subsystem. There are two redundant trains in the Safeguards Area Ventilation subsystem. Each train of the Safeguards Area Ventilation subsystem consists of one Safeguards Area exhaust fan, prefilter, and high efficiency particulate air (HEPA) filter and charcoal adsorber assembly for removal of gaseous activity (principally iodines) (shared with the other unit), and controls for the Safeguards Area exhaust filter and bypass dampers. Ductwork, valves or dampers, and instrumentation also form part of the subsystem. The subsystem automatically initiates filtered ventilation of the safeguards pump room following receipt of a Containment Hi-Hi signal from the affected unit. 8

The Auxiliary Building Central exhaust subsystem consists of the following: three redundant central area exhaust fans (shared with other unit), two redundant filter banks consisting of HEPA filter and charcoal adsorber assembly for removal of gaseous activity (principally iodines) (shared with the other unit), and two redundant trains of controls for the Auxiliary Building Central exhaust subsystem filter and bypass dampers (shared with the other unit). Ductwork, valves or dampers, and instrumentation also form part of the subsystem. The subsystem initiates filtered ventilation of the charging pump cubicles following manual actuation. 8 R13

The Auxiliary Building filter banks are shared by the Safeguards Area Ventilation subsystem and the Auxiliary Building Central Exhaust subsystem. Either Auxiliary Building filter bank may be aligned to either ECCS PREACS train. These filter banks are also used by the Auxiliary Building General area exhaust, fuel building exhaust, decontamination building exhaust, and containment purge exhaust. 6

One Safeguards Area exhaust fan is normally operating and dampers are aligned to bypass the HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the Safeguards Area room are diverted through the filter banks. Two Auxiliary Building Central Exhaust fans are normally operating. Air discharges from the Auxiliary Building Central Exhaust area are manually diverted through the filter banks. Required Safeguards Area and Auxiliary Building Central Exhaust area fans are manually actuated if they are not already operating. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

BASES

during normal operations

BACKGROUND (continued)

heaters is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. ~~4~~ 3). generally The heaters are not required for post-accident conditions (2) |R13

APPLICABLE SAFETY ANALYSES

The design basis of the ~~ECCS PREACS~~ leakage is established by the large break LOCA. The system evaluation assumes ~~a~~ passive failure of the ECCS outside containment, such as ~~an~~ Safety Injection ~~SI~~ pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 ~~CFR 100 (Ref. 5) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of Reference 5 limits)~~. The analysis of the effects and consequences of a large break LOCA is presented in Reference ~~3~~ 2. The ECCS PREACS also actuates following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing. (9) (2) (2) (2) |R13

control room operator dose limits of 10CFR50, Appendix A, GDC-19 (Ref. 4), and NUREG-0800, Section 6.4

may

The analyses assume that filtration by the ECCS PREACS does not begin for 60 minutes following an accident.

~~Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.~~ (13)

The ECCS PREACS satisfies Criterion 3 of ~~the NRC Policy Statement~~. 10CFR50.36(c)(2)(ii) (5)

LCO

Two ~~independent and~~ redundant trains of the ECCS PREACS are required to be OPERABLE to ensure that at least one is available, ~~assuming that a single failure disables the other train coincident with loss of offsite power~~. Total system failure could result in the atmospheric release from the ~~ECCS pump room exceeding 10 CFR 100 limits in the event of a Design Basis Accident (DBA)~~. (8) (8) (2)

exceeding the control room operator dose limits of 10CFR50, Appendix A, GDC-19 (Ref. 4), and NUREG-0800, Section 6.4 (Ref. 5)

ECCS PREACS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration are OPERABLE in both trains.

An ECCS PREACS train is considered OPERABLE when its associated:

(continued)

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BASES INSERT 4 (4)

LCO (continued)

(2)

a. ~~Fan is OPERABLE;~~ ← Insert 1

(e) ~~b.~~ HEPA filter and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

(2)

(f) ~~c.~~ ~~Heater, demister,~~ ductwork, valves, and dampers are OPERABLE ~~and air circulation can be maintained~~.

INSERT 2 → TSTF-287 (4)

APPLICABILITY

In MODES 1, 2, 3, and 4, the ECCS PREACS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS. Insert 5 (14) | R13

In MODE 5 or 6, the ECCS PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With one ECCS PREACS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ECCS PREACS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability. Design Basis Accident (DBA) (2)

Concurrent failure of two ECCS PREACS trains would result in the loss of functional capability; therefore, LCO 3.0.3 must be entered immediately.

INSERT 3 → TSTF-287

C
~~B~~.1 and ~~B~~.2 (s) or ECCS pump room boundary (3) | R13

If the ECCS PREACS train(s) or ECCS pump room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least

(continued)

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INSERT 1

a. Safeguards Area exhaust fan is OPERABLE;
b. One Auxiliary Building HEPA filter and charcoal adsorber assembly (shared with the other unit) is OPERABLE; R13
c. One Auxiliary Building Central exhaust system fan (shared with other unit) is OPERABLE; R13
d. Controls for the Auxiliary Building Central exhaust system filter and bypass dampers (shared with the other unit) are OPERABLE; R13

INSERT 2

The LCO is modified by a Note allowing the ECCS pump room boundary openings not open by design to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for ECCS pump room isolation is indicated.

INSERT 3

B.1

If the ECCS pump room boundary is inoperable, the ECCS PREACS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ECCS pump room boundary within 24 hours. During the period that the ECCS pump room boundary is inoperable, appropriate compensatory measures consistent with the intent of GDC 19 should be utilized to protect control room operators from potential hazards such as radioactive contamination. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump room boundary.

INSERT 4

In addition, the required Safeguards Area and charging pump cubicle boundaries for charging pumps not isolated from the RCS must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, except for those openings which are left open by design, including charging pump cubicle ladder wells.

INSERT 5

The Auxiliary Building Central Exhaust subsystems may be removed from service (e.g., tag out fans, open ductwork, etc.), in order to perform required testing and maintenance. The Auxiliary Building Central Exhaust subsystem is OPERABLE in this condition if it can be restored to service and perform its function within 60 minutes following an accident. R13

BASES

TSTF-287

ACTIONS

C

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

The system

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

and HEPA filters 12

1

INSERT 3

3

SR 3.7.12.2 3

This SR verifies that the required ECCS PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The ECCS PREACS filter tests are in accordance with Reference 4. The [VFTP] includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum and maximum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the [VFTP].

1

7 6

1

10

1

3

SR 3.7.12.3 4

diverts its exhaust flow through the filters

2

This SR verifies that each ECCS PREACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 4 3.

1

2

(continued)

Rev. 0

INSERT

SR 3.7.12.2

This SR verifies that Safeguards Area exhaust flow and Auxiliary Building Central Exhaust subsystem flow, when actuated from the control room, diverts flow through the Auxiliary Building HEPA filter and charcoal adsorber assembly for the operating train. Exhaust flow is diverted manually through the filters in case of a DBA requiring their use. The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available. R13

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.~~4~~ 5

This SR verifies the integrity of each train of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested in a qualitative manner to verify proper functioning of the ECCS PREACS. During the [post accident] mode of operation, the ECCS PREACS is designed to maintain a slight negative pressure in the ECCS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ECCS PREACS is designed to maintain a negative pressure ~~≤ [-0.125] inches water gauge~~ relative to adjacent areas ~~atmospheric pressure at a flow rate of [3000] cfm from the ECCS pump room~~. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. ~~6~~ 5).

This test is conducted with the tests for filter penetration; thus, an [18] month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference ~~4~~ 3.

~~SR 3.7.12.5~~

~~Operating the ECCS PREACS bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the ECCS PREACS bypass damper is verified if it can be specified in Reference 4.~~

REFERENCES

~~1. FSAR, Section [6.5.1].~~

~~2~~ 1. FSAR, Section [9.4.~~5~~].

~~3~~ 2. FSAR, Section [15.6.~~5~~].

~~4~~ 3. Regulatory Guide 1.52 (Rev. 2).

~~5~~ 4. ~~10 CFR 100.11~~ 10 CFR 50, Appendix A.

~~6~~ 5. NUREG-0800, ~~Section 6.5.1,~~ Rev. 2, July 1981.

Rev. 13

10. Testing of the maximum flow rate is added to the testing of the activated charcoal listed in the Bases for ITS SR 3.7.12.3 as part of the Ventilation Filter Testing Program. Adding the maximum flow rate is consistent with the Section 5.0 discussion of the VFTP. The maximum flow rate is an appropriate test criteria because of residence times associated with the activated charcoal.

11. ISTS SR 3.7.12.4 Bases are modified to state that the surveillance is performed in a qualitative manner. ISTS SR 3.7.12.4 is modified to require one ECCS PREACS train maintain a negative pressure relative to atmospheric pressure during post accident mode of operation, and does not specify a specific pressure or flow rate. The Safeguards Area and Auxiliary Building Central exhaust area are not maintained at a specific negative pressure due to the design of these areas. Also, a specific negative pressure is not assumed as part of the DBA analysis, and the ECCS PREACS flow rate is verified as part of the Ventilation Filter Testing Program.

12. The Bases for ITS SR 3.7.12.1 are modified to state that monthly heater operations dry out moisture in the HEPA filters in addition to the charcoal. Drying out the HEPA filters is also an important result of the surveillance, and is added for clarification.

13. A discussion in the Applicable Safety Analyses section system regarding failures considered in the accident analysis is deleted. The discussion concerns a complete loss of function and excessive LEAKAGE, two assumptions which are actually beyond the analysis. The analysis assumes LEAKAGE within assumed limits, and that at least one train of the system functions. This paragraph is not consistent with DBA analysis.

14. The ECCS PREACS trains consist of two subsystems; the Safeguards Area Ventilation subsystem and the Auxiliary Building Central Exhaust subsystem. The description of the ECCS PREACS is revised and expanded to discuss each subsystem separately. The Auxiliary Building Central Exhaust subsystem is shared between Units 1 and 2 and both trains share common ductwork. As a result, any opening of the ductwork could be construed as rendering both trains of the Auxiliary Building Central Exhaust subsystem inoperable for both units. This would prevent any maintenance, repair or testing on the Auxiliary Building Central Exhaust subsystem unless both units are shutdown. This is inconsistent with current operating procedures. The Auxiliary Building Central Exhaust subsystem does not appear in the CTS. The accident analyses assume that filtration by the ECCS PREACS does not begin for 60 minutes following an accident. Plant procedures allow the Auxiliary Building Central Exhaust subsystem to be considered OPERABLE during maintenance, testing and repair provided the subsystem can be restored to service following an accident before the ECCS PREACS is needed. That interpretation is incorporated into the ITS Bases. R13

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS SR 3.8.1.4 Bases

Description

The Bases of SR 3.8.1.4 are revised to state that the SR verifies that the required amount of fuel oil is in the day tank, not that the fuel oil level is at or above the level at which fuel oil is automatically added, because the setpoint is set above the volume required to meet the SR. This affects the ITS Bases, the ISTS Bases markup, and adds JFD 11.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is at or above the level which is required. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of EDG operation at full load plus 10%. R13

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and operators would be aware of any large uses of fuel oil during this period. R3

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 92 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during EDG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are consistent with the recommendations of Regulatory Guide 1.137 (Ref. 9). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for fuel transfer systems are OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source. E 1

E Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between [0.8 lagging] and [1.0]. The [0.8] value is the design rating of the machine, while the [1.0] is an operational limitation [to ensure circulating currents are minimized]. The load band is provided to avoid routine E overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. E 1 2 2 1

TSTF 37

The 31 day Frequency for this Surveillance [Table 3.8.1-1] is ~~consistent with Regulatory Guide 1.9 (Ref. 3)~~ acceptable based on operating experience. 1

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A E successful DG start must precede this test to credit satisfactory performance. E 1 1

SR 3.8.1.4

required

This SR provides verification that the level of fuel oil in the day tank [and engine mounted tank] is at or above the level at which [fuel oil] is [automatically added]. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour E of DG operation at full load plus 10%. 2 11 RB 1

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are

(continued)

Rev. 13

11. The Bases of SR 3.8.1.4 are revised to reflect the plant design. SR 3.8.1.4 states, "Verify each required day tank contains ≥ 450 gal of fuel oil." The ISTS states, "This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added." This statement is revised to state, "This SR provides verification that the level of fuel oil in the day tank is at or above the level which is required." The ISTS Bases information is not accurate for North Anna because the setpoint for automatically filling the day tank is set above the 450 gallon requirement in the SR. R13

12. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS SR 3.8.1.7

Description

A typographical error is corrected in SR 3.8.1.7. The SR previously stated, "Verify each required EDG starts from standby condition and achieves in." The word "in" is deleted from the SR. The ISTS Bases markup is correct.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.4	Verify each required day tank contains ≥ 450 gal of fuel oil.	31 days
SR 3.8.1.5	Check for and remove accumulated water from each required day tank.	92 days
SR 3.8.1.6	Verify each required fuel oil transfer pump operates to transfer fuel oil from the storage tank to the day tank.	92 days
SR 3.8.1.7	-------------------NOTE------------------- All EDG starts may be preceded by an engine prelube period. -- Verify each required EDG starts from standby condition and achieves a. In ≤ 10 seconds, voltage ≥ 3960 V and frequency ≥ 59.5 Hz; and b. Steady state voltage ≥ 3740 V and ≤ 4580 V, and frequency ≥ 59.5 Hz and ≤ 60.5 Hz.	184 days R13
SR 3.8.1.8	-------------------NOTES------------------- 1. This Surveillance is only applicable to Unit 1. 2. This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the unit is maintained or enhanced. RAI 3.8.1-02 R3 -- Verify manual transfer of AC power sources from the normal offsite circuit to the alternate required offsite circuit.	18 months

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS SR 3.8.1.15 Bases

Description

The Bases of SR 3.8.1.15 says the Surveillance ensures that "the manual synchronization and automatic load transfer from the EDG to the offsite source can be made..." At North Anna, the transfer of load from an EDG to the offsite source is made manually. The ITS Bases are revised to delete the word "automatic." This affects the ITS Bases and ISTS Bases markup.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

do not invalidate this test. Note 2 allows all EDG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.15

Consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and load transfer from the EDG to the offsite source can be made and the EDG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the EDG to reload if a subsequent loss of offsite power occurs. The EDG is considered to be in ready to load status when the EDG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequencing timing relays are reset. EDG loading of the emergency bus is limited to normal energized loads.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1, 2, 3, or 4 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a unit shutdown and startup to determine that unit safety

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.16 5

Consistent with the recommendations of ~~As required by~~ Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timers are reset. ing ing relays <INSERT 1>

5

7 / R3 9

E 1 / R13

1

1

2

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 8), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance. 8 9

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. ~~Credit may be taken for unplanned events that satisfy this SR.~~ <INSERT 2> TSTF 283 10 TSTF 8

~~SR 3.8.1.17~~

~~Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready to load operation if a LOCA actuation signal is received during operation in the test mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 13), paragraph 6.2.6(2).~~

~~The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable.~~

5

(continued)

Rev 13

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS 3.9.1 Bases

Description

The Applicable Safety Analysis Bases of ITS 3.9.1 is revised to clarify the purpose of the LCO. The Bases state that the reactivity condition of the core is established to protect against inadvertent reactivity addition. This is incorrect. The reactivity condition of the core is established to protect against inadvertent positive reactivity additions. The ITS Bases and ISTS markup are revised to state positive reactivity additions.

The Bases of SR 3.9.1.1 state that the boron concentration of the coolant "in each volume" is determined periodically by chemical analysis. This implies that it is necessary to always sample the RCS, the refueling canal, and the refueling cavity separately to perform the SR. The refueling canal and the refueling cavity represent a single volume which is filled and cooled from the RCS. Therefore, in most circumstances, three separate samples will not be required to verify the boron concentration is within the limit. The phrase "in each volume" is deleted from the SR 3.9.1.1 Bases. This affects the ITS Bases and the ISTS markup. JFD 6 is added to describe the change.

BASES

BACKGROUND (continued)

refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is established to protect against inadvertent positive reactivity addition and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance. R13

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6,

(continued)

BASES

ACTIONS

A.3 (continued)

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, and connected portions of the refueling canal and the refueling cavity, is within the COLR limits. The boron concentration of the coolant is determined periodically by chemical analysis. Prior to re-connecting portions of the refueling canal or the refueling cavity to the RCS, this SR must be met per SR 3.0.1. If any dilution activity has occurred while the cavity or canal were disconnected from the RCS, this SR ensures the correct boron concentration prior to communication with the RCS. R13

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. UFSAR, Section 3.1.22.

BASES

BACKGROUND (continued)

canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is ~~consistent with the initial conditions assumed for the boron dilution accident in the accident analysis~~ established to protect against inadvertent positive reactivity addition and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance. (2) | R13

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

~~The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.2, "SHUTDOWN MARGIN (SDM)—T_{avg} $\leq 200°F$."~~ (2)

The RCS boron concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). (3)

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of

(continued)

Rev. 13

BASES

ACTIONS

A.3 (continued)

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

and connected portions of

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis. Insert 6 TSTF-272/RL

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26.~~ 1

~~2. FSAR, Chapter [15].~~ 2

UFSAR Section 3.1.22.

Rev. 13

1. North Anna Units 1 and 2 were designed and constructed on the basis of the proposed General Design Criteria, published in 1966. Since February 20, 1971, when the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50, were published, the Company attempted to comply with the intent of the newer criteria to the extent practical, recognizing previous design commitments. The NRC's Safety Evaluation Report for North Anna Units 1 and 2 reviewed the plant against 10 CFR Part 50, Appendix A and concluded that the facility design conforms to the intent of the newer criteria. The North Anna UFSAR contains discussions comparing the design of the plant to the 10 CFR 50, Appendix A, General Design Criteria. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

2. The Bases are revised to reflect the North Anna boron dilution analysis. The North Anna analysis is based on locking out the primary grade water sources. As a result, there is no "limiting" boron dilution analysis. A detailed discussion of this event does not appear in the Bases for Specification 3.1.1. Therefore, these sentences are deleted.

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. 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.

5. Editorial changes are made to the Bases to be consistent with the ITS or to make the sentences grammatically correct.

6. The Bases of SR 3.9.1.1 are revised. The Bases state that the boron concentration of the coolant "in each volume" is determined periodically by chemical analysis. This implies that it is necessary to always sample the RCS, the refueling canal, and the refueling cavity separately to perform the SR. The refueling canal and the refueling cavity represent a single volume which is filled and cooled from the RCS. Therefore, in most circumstances, three separate samples will not be required to verify the boron concentration is within the limit. The phrase "in each volume" is deleted from the SR 3.9.1.1 Bases. R13

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

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

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

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the ECST to each steam generator prior to entering MODE 3 after more than 30 contiguous days in any combination of MODES 5, 6, or defueled. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the ECST to the steam generators is properly aligned.

REFERENCES

1. UFSAR, Section 10.4.3.2.

2. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

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

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

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of ~~centrigufal~~ centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref 2). Because it is sometimes undesirable to introduce cold AFW into the steam generators while they are operating, this testing is typically performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code ~~Section XI~~ (Ref. 2) (only required at 3 month intervals) satisfies this requirement. ~~The [31] day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.~~ 6/R13 2 6/R13 TSTF-101

[This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there ~~is~~ may be insufficient steam pressure to perform the test.] 1 4

(continued)

Rev. 13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.4 (continued)

[Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.] ①

[SR 3.7.5.5

Ⓔ contiguous

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam ③ generator prior to entering MODE ② after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has Ⓔ occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. (This SR is not required by those units that use AFW for normal startup and shutdown.)]

any combination of MODE 5 or 6 or defueled.

TSTF-268 ④

①

④

REFERENCES

1. ④ FSAR, Section [10.4.9]. 3.7 ④

2. ASME, Boiler and Pressure Vessel Code, Section XI. ⑥ RB

ASME Code for Operation and Maintenance of Nuclear Power Plants.

Rev. 13

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

2. An editorial change is made for clarity, for consistency with the Improved Technical Specifications Writer's Guide, or for consistency with similar statements in the other ITS Bases.

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. Changes have been made (additions, deletions, or changes to the NUREG-1431) to reflect the facility-specific nomenclature, number, reference, system description, or analysis description.

5. This change to the Bases is necessary for consistency with the specification.

6. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

The 92 day Frequency corresponds to the testing requirements of pumps as contained in the ASME Code (Ref. 10). The fuel oil transfer system is such that the pumps must be started manually in order to maintain an adequate volume of fuel in the day tank during or following EDG testing, and a 92 day Frequency is appropriate. R13

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads for Unit 1 only. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. Note 1 states that the SR is applicable to Unit 1 only. The SR is not applicable to Unit 2 because it does not have an alternate offsite feed for the emergency buses. The reason for Note 2 is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines unit safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite RAI 3.8.1-02 R3 RAI 3.8.1-02 R3

(continued)

BASES

REFERENCES (continued)

8. Regulatory Guide 1.108, Rev. 1, August 1977.
9. Regulatory Guide 1.137, Rev. 1, October 1979.
10. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13
11. IEEE Standard 308-1971.
12. Technical Requirements Manual.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

7

12 1R13

the the 6

[Section XI (Ref. 10); however, the design of fuel transfer systems is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day [and engine mounted] tanks during or following DG testing. In such a case, a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.]

2

E and

SR 3.8.1.7

See SR 3.8.1.2.

[SR 3.8.1.8

Transfer of each [4.16 kV ESF bus] power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The [18 month] Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

for Unit 1 only 5

2

<INSERT 1> 5

Two 5 2

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Credit may be taken for unplanned events that satisfy this SR.]

RAI 3.8.1 02 R3

TSTF-8

<INSERT 2> TSTF 283

RAI 3.8.1 02 R3

SR 3.8.1.9

E

1

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine

(continued)

Rev 13

BASES

REFERENCES (continued)

8 ~~9~~. Regulatory Guide 1.108, Rev. 1, August 1977. ⑨

9 ~~10~~. Regulatory Guide 1.137, Rev. ~~[ ], [date]~~ 1, October 1979 ② ⑨

10 ~~11~~. ASME, ~~Boiler and Pressure Vessel Code, Section XI~~ ⑬ IR13 ① ⑨

ASME Code for Operation and Maintenance of Nuclear Power Plants.

11 ~~12~~. IEEE Standard 308-197~~8~~ ① ① ⑨

12. Technical Requirements Manual

Rev. 13

11. The Bases of SR 3.8.1.4 are revised to reflect the plant design. SR 3.8.1.4 states, "Verify each required day tank contains ≥ 450 gal of fuel oil." The ISTS states, "This SR provides verification that the level of fuel oil in the day tank is at or above the level at which fuel oil is automatically added." This statement is revised to state, "This SR provides verification that the level of fuel oil in the day tank is at or above the level which is required." The ISTS Bases information is not accurate for North Anna because the setpoint for automatically filling the day tank is set above the 450 gallon requirement in the SR. R13

12. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS 5.5.10

Description

The ISTS NUREG Ventilation Filter Testing Program (VFTP) contains an optional requirement for testing heaters on ESF systems. There is not a requirement to test heaters on ESF systems in the CTS. It has been determined that the heaters on the Main Control Room (MCR) / Emergency Switchgear Room (ESGR) Emergency Ventilation System (EVS) are required to support MCR/ESGR EVS OPERABILITY. Therefore, a requirement is added to the VFTP to test those heaters. This affects the ITS, the ISTS markup, the Unit 1 and Unit 2 CTS markup, and adds DOC M.26.

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

c. (continued)

value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and relative humidity specified below.

ESF Ventilation System	Penetration	RH
MCR/ESGR EVS	2.5%	70%
ECCS PREACS	5%	70%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1975 at the system flowrate specified below. RAI 5.0-06 R4 RAI 5.0-06 R4

ESF Ventilation System	Delta P	Flowrate
MCR/ESGR EVS	4 inches W.G.	1000 ± 10% cfm
ECCS PREACS	5 inches W.G.	≤ 39,200 cfm

R13

e. Demonstrate that the heaters for each of the ESF systems dissipate ≥ the value specified below when tested in accordance with ASME N510-1975.

ESF Ventilation System	Wattage
MCR/ESGR EVS	3.5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or RAI 5.0-07 R4

(continued)

CTS

5.5 Programs and Manuals

4.7.7.1.d 5.5.~~11~~ 10 Ventilation Filter ANSI Testing Program (VFTP) (continued) 7 1 RAI 5.0-06 R4

Revision 2 and ~~ASME~~ N510-~~1989~~ 1975 at the system flowrate specified below ~~[± 10%]~~.

ESF Ventilation System	Delta P	Flowrate
MCR/ESGR EVS	4 inches W.G.	1000 cfm ±10%
ECCS PREACS	5 inches W.G.	≤ 39,200 cfm

1

e. Demonstrate that the heaters for each of the ESF systems dissipate ≥ the value specified below ~~[± 10%]~~ when tested in accordance with ANSI N510-~~1989~~ 1975. 1 R13

ESF Ventilation System	Wattage
MCR/ESGR EVS	3.5 Kw

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

New 5.5.~~12~~ 11 Explosive Gas and Storage Tank Radioactivity Monitoring Program 7 1

This program provides controls for Gaseous Waste ~~potentially explosive gas mixtures contained in the Waste Gas Holdup System~~, the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system~~, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures". 1 2 1 RAI 5.0-7 R4 1 1

The program shall include:

-3.11.2.5 a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste ~~Waste Gas Holdup~~ System and a surveillance program to ensure the limits are maintained. Such limits shall be 1 RAI 5.0-7 R4

(continued)

Rev. 13

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an acceptable residence time within the charcoal adsorber.

M.21

ITS

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Cont'd)

5.5.10.c 2. Verifying, ~~within 31 days after removal,~~ that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70 ~~95~~%. LA.5 L.33

5.5.10.a 5.5.10.b 3. Verifying ~~a system~~ flow rate ~~of 6,300 cfm ± 10% during operation~~ of one ECCS PREACS train provides greater than the minimum required cooling flow for ECCS equipment when tested in accordance with ANSI N510-1975. M.21

5.5.10.c c. ~~Within 31 days of completing 720 hours of charcoal adsorber operation,~~ verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of ~~95~~% 70. LA.5 L.33

d. ~~At least once per 18 months by:~~ LA.5

5.5.10.d 1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is < ~~6~~ 5 inches Water Gauge while operating the ~~ventilation system~~ ECCS PREACS at a flow rate of ~~6,300 cfm ± 10%~~ ≤ 39,200 cfm. M.21

2. Verifying that on a Containment Hi-Hi Test Signal, the system automatically diverts its exhaust flow through the auxiliary building HEPA filter and charcoal adsorber assembly. See ITS 3.7.12

5.5.10.a e. ~~After each complete or partial replacement of a HEPA filter bank by~~ verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating ~~the system at a flow rate of 6,300 cfm ± 10%~~. LA.5 M.21

5.5.10.b f. ~~After each complete or partial replacement of a charcoal adsorber bank by~~ verifying that that charcoal adsorbers remove ≥ 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating ~~the system at a flow rate of 6,300 cfm ± 10%~~. LA.5

one ECCS PREACS train at nominal accident flow M.21

5.5.10.e Insert ITS 5.5.10.e M.26 R13

Nominal accident flow for a single train activation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤39,200 cfm, which is the maximum flow rate providing an acceptable residence time within the charcoal adsorber. (M.21)

SURVEILLANCE REQUIREMENTS (cont'd)

ITS

5.5.10.c 2. Verify~~ing, within 31 days after removal,~~ that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of ~~95~~ 70%. (LA.5) (L.33)

5.5.10.a 5.5.10.b 3. Verifying ~~a system~~ of one ECCS PREACS train provides greater than the minimum required cooling flow for ECCS equipment flow rate ~~of 6,300 cfm ± 10% during operation~~ when tested in accordance with ANSI N510-1975. (M.21)

5.5.10.c c. ~~Within 31 days of completing 720 hours of charcoal adsorber operation,~~ verify that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than or equal to 5% when tested in accordance with ASTM D 3803-1989 at a temperature of 30°C (86°F) and a relative humidity of ~~95~~ 70%. (LA.5) (L.33)

d. ~~At least once per 18 months by:~~ (LA.5)

5.5.10.d 1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is less than ~~6~~ 5 inches Water Gauge while operating the ~~ventilation system~~ ECCS PREACS at a flow rate of ~~6,300 cfm ± 10%~~ ≤ 39,200 cfm. (M.21)

2. Verifying that on a Containment Pressure–High-High Test Signal, the system automatically diverts its exhaust flow through the auxiliary building HEPA filter and charcoal adsorber assembly. (See ITS 3.7.12)

5.5.10.a e. ~~After each complete or partial replacement of a HEPA filter bank by~~ verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating ~~the system at a flow rate of 6,300 cfm ± 10%~~. (LA.5) (M.21)

5.5.10.b f. ~~After each complete or partial replacement of a charcoal adsorber bank by~~ verifying that that charcoal adsorbers remove greater than or equal to 99% of a halgenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating ~~the system at a flow rate of 6,300 cfm ± 10%~~ One ECCS PREACS train at nominal accident flow. (LA.5) (M.21)

5.5.10.e Insert ITS 5.5.10.e (M.26) RB

The purpose of the AO requirements in CTS Table 6.2-1 is to provide assurance that sufficient AOs are on the shift crew. This change is acceptable because it still provides at least three AOs with both units shutdown or defueled. This change is designated more restrictive because an additional AO is required. R4

M.25 ITS 5.6.5.b contains two analytical methods, WCAP-8745-P-A and WCAP-14483-A, which do not appear in the CTS. This changes the CTS by adding two analytical methods to those referenced in the Technical Specifications. MB2073 MB2075 R11

The purpose of the analytical methods referenced in 5.6.5 is to provide the NRC approved methodologies used to determine values in the COLR. Changes justified in other ITS Sections have relocated values to the COLR. These two analytical methods are used to determine those values. This change is designated as more restrictive because additional analytical methods are listed in the Technical Specifications.

M.26 ITS 5.5.10.e requires testing of MCR/ESGR EVS heaters to verify they dissipate the required wattage. The CTS does not contain this requirement. This changes the CTS by adding a Surveillance Requirement. R13

The purpose of the test is to verify that the MCR/ESGR EVS heaters can perform their required function. This change is acceptable because it provides appropriate testing to verify system OPERABILITY. This change is designated as more restrictive because additional testing is required.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 6.8.1.i requires written procedures be established, implemented and maintained covering, "Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975." ITS 5.4.1.c does not include the Regulatory Guide references. This changes the CTS by moving the references to the Regulatory Guides to the UFSAR.

The removal of these details for performing actions 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 for procedures covering quality assurance for effluent and environmental monitoring. Also, this change is acceptable because these types of procedural details will be adequately controlled in the UFSAR. The UFSAR is controlled under 10 CFR 50.59 which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because references for meeting Technical Specification requirements are being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS 5.7.1 states, "The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1." CTS Table 5.7-1 contains the limits for component cyclic or transient limits and designs cycle or transient limits. ITS 5.5.5 states, "The components identified in the UFSAR, Section 5.2, are designed and shall

Specifications Affected: ITS Chapter 5.0

<u>Description</u>

The page numbers of ITS Chapter 5.0 are revised to use a section-based page number, consistent with Revision 2 of NUREG-1431. A complete copy of ITS Chapter 5.0 is provided.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 The Shift Supervisor (SS) shall be responsible for the control room command function. During any absence of the SS from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

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5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications shall be documented in the UFSAR/QA Plan;

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2 Organization

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. An auxiliary operator shall be assigned to each reactor containing fuel and an additional auxiliary operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. R4

   Two unit sites with both units shutdown or defueled require a total of three auxiliary operators for the two units.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), health physicists, auxiliary operators, and key maintenance personnel).

   The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime.

   Any deviation from the above guidelines shall be authorized in advance by the plant manager or the plant manager's designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

   Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.

5.2 Organization

5.2.2 Unit Staff (continued)

e. The operations manager shall hold (or have previously held) a Senior Reactor Operator License for North Anna or a similar design Pressurized Water Reactor plant. The Supervisor Shift Operations shall hold an active Senior Reactor Operator License for North Anna Power Station. R4

f. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

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5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI 3.1 (12/79 Draft) for comparable positions. Exceptions to this requirement are specified in VEPCO's QA Topical Report, VEP-1, "Quality Assurance Program, Operational Phase." The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The SS, Assistant SS, Control Room Operator-Nuclear, and the individual providing advisory technical support to the unit operations shift crew, shall meet or exceed the minimum qualifications of 10 CFR 55.59(c) and 55.31(a)(4). R4

5.3.2 For the purpose of 10 CFR 55.4, a licensed SRO and a licensed RO are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

Intentionally Blank

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5.

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5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Annual Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after the approval of the plant manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 18 months. R4

The provisions of SR 3.0.2 are applicable. R13

5.5.3 Reserved

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402; R13

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days; R13

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. For noble gases: a dose rate $\leq$ 500 mrem/yr to the whole body and a dose rate $\leq$ 3000 mrem/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate $\leq$ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency. R4

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR, Section 5.2, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel once every 10 years by a qualified inplace UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces defined by the volume of disassembled flywheels.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program surveillance frequency.

5.5 Programs and Manuals

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in the ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda as follows: R13

ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

R13

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Code for Operation and Maintenance of Nuclear Power Plants shall be construed to supersede the requirements of any TS. R13

5.5.8 Steam Generator (SG) Tube Surveillance Program

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test Frequencies.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)

This program provides the controls for the inservice inspection of steam generator tubes to ensure that the structural integrity of this portion of the RCS is maintained. The program for inservice inspection of steam generators is based on a modification of Regulatory Guide 1.83, Revision 1. This program shall include:

5.5.8.1 Steam Generator Sample Selection and Inspection

Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5.8-1.

5.5.8.2 Steam Generator Tube Sample Selection and Inspection

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.8-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.8.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.8.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations > 20%, and

   2. Tubes in those areas where experience has indicated potential problems.

   3. A tube inspection (pursuant to Specification 5.5.8.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5.5 Programs and Manuals

5.5.8.2 Steam Generator Tube Sample Selection and Inspection (continued)

c. The tubes selected as the second and third samples (if required by Table 5.5.8.2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where, tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results[a]
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

a. In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the above percentage calculations.

5.5.8.3 Inspection Frequencies

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions,

(continued)

5.5 Programs and Manuals

5.5.8.3 Inspection Frequencies

a. (continued)

not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5.8-2 at 40 month intervals fall into category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.8.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A major steam line or feedwater line break.

5.5.8.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

5.5 Programs and Manuals

5.5.8.4 Acceptance Criteria

a. (continued)

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections > 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.8.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely around the U-bend to the top support.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy-current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed using the equipment and techniques expected to be used during subsequent inservice inspection.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.8-2.

Table 5.5.8-1
Minimum Number of Steam Generators to Be Inspected
During Inservice Inspection

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection		All		One	Two	Two
Second & Subsequent Inservice Inspection		One[1]		One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the unit) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

Table 5.5.8-2
Steam Generator Tube Inspection

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per SG	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in SG	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in SG	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this SG, plug defective tubes and inspect 2S tubes in each other SG	All other SGs are C-1	None	N/A	N/A
			Some SGs C-2 but no additional SG are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug defective tubes	N/A	N/A

S = 3[N/n]% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection. R4

5.5 Programs and Manuals

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include: RAI 5.0-05 R4

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems in general conformance with the frequencies and requirements of Regulatory Positions C.5.a, C.5.c, C.5.d, and C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

a. (continued)

with Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System (EVS)	1000 ± 10% cfm
Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)	Nominal accident flow for a single train actuation

R4

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an adequate residence time within the charcoal adsorber.

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Positions C.5.a and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
MCR/ESGR EVS	1000 ± 10% cfm
ECCS PREACS	Nominal accident flow for a single train actuation

Nominal accident flow for a single train actuation is greater than the minimum required cooling flow for ECCS equipment operation, and ≤ 39,200 cfm, which is the maximum flow rate providing an adequate residence time within the charcoal adsorber.

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than the

(continued)

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (VFTP)

c. (continued)

value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and relative humidity specified below.

ESF Ventilation System	Penetration	RH
MCR/ESGR EVS	2.5%	70%
ECCS PREACS	5%	70%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1975 at the system flowrate specified below. RAI 5.0-06 R4 RAI 5.0-06 R4

ESF Ventilation System	Delta P	Flowrate
MCR/ESGR EVS	4 inches W.G.	1000 ± 10% cfm
ECCS PREACS	5 inches W.G.	≤ 39,200 cfm

e. Demonstrate that the heaters for each of the ESF systems dissipate ≥ the value specified below when tested in accordance with ASME N510-1975. R13

ESF Ventilation System	Wattage
MCR/ESGR EVS	3.5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or RAI 5.0-07 R4

(continued)

5.5 Programs and Manuals

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); RAI 5.0-07 R4

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in each of the following outdoor tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains liquid radwaste ion exchanger system is less than the amount that would result in concentrations greater than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, excluding tritium, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents:

    1. Refueling Water Storage Tank;

    2. Casing Cooling Storage Tank;

    3. PG Water Storage Tank;

    4. Boron Recovery Test Tank; and

    5. Any Outside Temporary Tank.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals (continued)

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and R4

   3. water and sediment ≤ 0.05%.

b. Within 31 days following addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil;

c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with ASTM D-2276, Method A-2 or A-3; and

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing Frequencies.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license; or

5.5 Programs and Manuals

5.5.13 Technical Specifications (TS) Bases Control Program (continued)

b. (continued)

2. a change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.14 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident R4

(continued)

5.5 Programs and Manuals

5.5.14 Safety Function Determination Program (SFDP) (continued)

analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.15 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 44.1 psig. The containment design pressure is 45 psig. RAI 5.0-08 R4

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

5.5 Programs and Manuals

5.5.15 Containment Leakage Rate Testing Program (continued)

d. Leakage Rate acceptance criteria are:

1. Prior to entering a MODE where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

   $\leq 0.60\ L_a$ for the Type B and Type C tests on a Maximum Path Basis and $\leq 0.75\ L_a$ for Type A tests.

   During operation where containment OPERABILITY is required, the containment leakage rate acceptance criteria are:

   $\leq 1.0\ L_a$ for overall containment leakage rate and $\leq 0.60\ L_a$ for the Type B and Type C tests on a Minimum Path Basis.

2. Overall air lock leakage rate testing acceptance criterion is $\leq 0.05\ L_a$ when tested at $\geq P_a$. R4

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

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

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

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

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements commensurate with the format in the ODCM. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Annual Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Annual Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR) | R4

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Safety Limits,

2. SHUTDOWN MARGIN, | R11

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) R4

a. (continued)

3. Moderator Temperature Coefficient,
4. Shutdown Bank Insertion Limits,
5. Control Bank Insertion Limits,
6. AXIAL FLUX DIFFERENCE limits, R11
7. Heat Flux Hot Channel Factor,
8. Nuclear Enthalpy Rise Hot Channel Factor,
9. Power Factor Multiplier, RAI 5.0-10 R4
10. Reactor Trip System Instrumentation - OTΔT and OPΔT Trip Parameters,
11. RCS Pressure, Temperature, and Flow DNB Limits, and RAI 5.0-10 R4
12. Boron Concentration.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. VEP-FRD-42, "Reload Nuclear Design Methodology."
2. WCAP-9220-P-A, "WESTINGHOUSE ECCS EVALUATION MODEL—1981 VERSION."
3. WCAP-9561-P-A, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS—SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL."
4. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."
5. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) R4

b. (continued)

6. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."

7. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY—REFERENCE CORE REPORT."

8. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

9. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code."

10. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

11. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function." MB2073 MB2075 R11

12. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. R4

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Nuclear Regulatory Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

d. (continued)

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the radiation protection shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

   2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

5.7 High Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

d. (continued)

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual

(continued)

5.7 High Radiation Area

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u>

f. (continued)

area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

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Specifications Affected: ITS 3.9.6 Bases

Description

The LCO Bases of ITS 3.9.6 is revised to insert a missing word. The Bases state, "... there is no draining operation to further reduce RCS water and ..." is revised to state, "... there is no draining operation to further reduce RCS water level and ..." This is consistent with TSTF-361. This revises the ITS Bases and the ISTS Bases markup.

BASES

LCO (continued)

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

c. Indication of reactor coolant temperature.

This LCO is modified by two Notes. Note 1 permits the RHR pumps to be removed from operation for ≤ 15 minutes when switching from one train to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10°F below saturation temperature. The Note prohibits boron dilution or draining operations when RHR forced flow is stopped. Note 2 allows one RHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing unit configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible. RAI 3.9.5-1 R4 R13

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

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation

(continued)

INSERT

This LCO is modified by two Notes. Note 1 permits the RHR pumps to be removed from operation for ≤ 15 minutes when switching from one train to another. The circumstances for removing both RHR pumps from operation are to be limited to situations when the outage time is short and the core outlet temperature is maintained > 10 °F below saturation temperature. The Note prohibits boron dilution or draining operations when RHR forced flow is stopped. Note 2 allows one RHR loop to be inoperable for a period of 2 hours provided the other loop is OPERABLE and in operation. Prior to declaring the loop inoperable, consideration should be given to the existing unit configuration. This consideration should include that the core time to boil is short, there is no draining operation to further reduce RCS water level and that the capability exists to inject borated water into the reactor vessel. This permits surveillance tests to be performed on the inoperable loop during a time when these tests are safe and possible.

6 RAI 3.9.5-1 R4

R13

Specifications Affected: ITS Chapter 5.0, ISTS 3.3.3 Bases

Description

North Anna license amendment 229 (Unit 1) and 210 (Unit 2), dated December 19, 2001, eliminated the Post Accident Sampling System requirements from the CTS. The change to the CTS is consistent with approved Traveler TSTF-366. This CTS change is incorporated into the ITS. Note that ITS Program 5.5.3 is marked "Reserved" in lieu of deleting the program and renumbering the programs that follow. This program number will be used in a future amendment to the ITS. This changes the Chapter 5.0 ITS, ISTS markup, and CTS markup. The ISTS 3.3.3 Bases markup is also revised to be consistent with TSTF-366.

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 18 months. R4

The provisions of SR 3.0.2 are applicable. R13

5.5.3 Reserved

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402; R13

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

CTS

5.5 Programs and Manuals

6.15.c 5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

6.8.4.a 5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include [Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner]. The program shall include the following: (1)

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at ~~refueling cycle intervals or less.~~ least once per [18] months. The provisions of SR 3.0.2 are applicable. TSTF-299 (1) R4

5.5.3 ~~Post Accident Sampling~~ Reserved.

~~This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:~~ TSTF-366 R13

~~a. Training of personnel;~~

~~b. Procedures for sampling and analysis; and~~

~~c. Provisions for maintenance of sampling and analysis equipment.~~

6.8.4.e 5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to

(continued)

ITS 5.0

A.1

ITS

b. ~~In-Plant Radiation Monitoring~~

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

~~(i) Training of personnel,~~

~~(ii) Procedures for monitoring, and~~

~~(iii) Provisions for maintenance of sampling and analysis equipment.~~

LA.3

5.5.9

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. This program shall include: M.22 RAI 5.0-05 R4

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off control point chemistry conditions, and A.35

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action. Rev. 13

d. Deleted

Rev. 13

A.1

ITS 5.0

ITS

b. ~~In-Plant Radiation Monitoring~~

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

~~(i) Training of personnel,~~

~~(ii) Procedures for monitoring, and~~

~~(iii) Provisions for maintenance of sampling and analysis equipment.~~

LA.3

5.5.9

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation and low pressure turbine disc stress corrosion cracking. This program shall include: M.22 RAI 5.0-05 R4

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage,

(iv) Procedures for the recording and management of data, A.35

(v) Procedures defining corrective actions for all off control point chemistry conditions, and

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

R.13

d. Deleted

BASES

ACTIONS

C.1 (continued)

of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur. ~~Condition C is modified by a Note that excludes hydrogen monitor channels.~~ (5)

~~D.1~~

~~Condition D applies when two hydrogen monitor channels are inoperable. Required Action D.1 requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA (which would cause core damage) would occur during this time.~~ TSTF-366 (5)/R13

~~E.1~~

~~Condition E applies when the Required Action and associated Completion Time of Condition C or D are not met. Required Action E.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C or D, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.~~ (5)

D.1 and D.2 (5)

If the Required Action (15) and associated Completion Time of Condition ~~s~~ C ~~or D~~ ~~are~~ not met ~~and Table 3.3.3-1 directs entry into Condition F,~~ the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours. (5)

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions

(continued)

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Specifications Affected: ITS Chapter 5.0

Description

Three typographical errors are corrected in ITS 5.5.4. ITS 5.5.4.b refers to 10 CFR 20.10001. The correct reference is to 10 CFR 20.1001. The spelling of the word "calendar" is corrected in two places in ITS 5.5.4.e. The ISTS markup was correct.

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 18 months. R4

The provisions of SR 3.0.2 are applicable. R13

5.5.3 Reserved

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402; R13

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days; R13

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

   1. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin, and

   2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

Specifications Affected: CTS Markup for ITS 5.5.8

Description

The CTS markup of Chapter 5.0, page 32 of 69 for both the Unit 1 and Unit 2 markup, incorrectly identified the Steam Generator Tube Surveillance Program as Specification 5.5.9. The correct specification number is 5.5.8. The markup is corrected.

The ITS and Bases are unaffected.

5.5.8 Steam Generator (SG) Tube Surveillance Program

ITS

This program provides the controls for the inservice inspection of steam generator tubes to ensure that the structural integrity of this portion of the RCS is maintained. The program for inservice inspection of steam generators is based on a modification of Regulatory Guide 1.83, Revision 1. This program shall include: A.38 R4

~~REACTOR COOLANT SYSTEM~~

~~STEAM GENERATORS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.4.5 Each steam generator in a non-isolated reactor coolant loop shall be OPERABLE.~~

See ITS 3.4.13

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

~~ACTION:~~

~~With one or more steam generators in non-isolated reactor coolant loops inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the required Specification 4.0.5.~~ A.7

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program Test Frequencies

5.5.8.1 4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table ~~4.4-1~~ 5.5.8-1.

5.5.8.2 4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table ~~4.4-2~~ 5.5.8-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification ~~4.4.5.3~~ 5.5.8.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification ~~4.4.5.4~~ 5.5.8.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except: A.20

5.5.8.2.a a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

5.5.8.2.b b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

A.1

ITS 5.0

5.5.8 Steam Generator (SG) Tube Surveillance Program

8-21-80

1R13

ITS

REACTOR COOLANT SYSTEM

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

This program provides the controls for the inservice inspection of steam generator tubes to ensure that the structural integrity of this portion of the RCS is maintained. The program for inservice inspection of steam generators is based on a modification of Regulatory Guide 1.83, Revision 1. This program shall include:

A.38 | R4

~~3.4.5 Each steam generator in a non-isolated reactor coolant loop shall be OPERABLE.~~

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

~~ACTION:~~

~~With one or more steam generators in non-isolated reactor coolant loops inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.~~

See ITS 3.4.13

~~SURVEILLANCE REQUIREMENTS~~

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program Test Frequencies

~~4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the required Specification 4.0.5.~~

A.7

5.5.8.1 4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table ~~4.4-1~~ 5.5.8-1.

5.5.8.2 4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table ~~4.4-2~~ 5.5.8-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification ~~4.4.5.3~~ 5.5.8.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification ~~4.4.5.4~~ 5.5.8.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

A.20

5.5.8.2.a a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

5.5.8.2.b b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

Specifications Affected: ITS Chapter 5.0 CTS markup and DOC

Description

ITS Chapter 5.0, DOCs A.21 and LA.11 state that the Inservice Inspection Program is included in the Inservice Testing Program as defined in ITS 5.5.7. This is incorrect. The Inservice Inspection Program is not included in the ITS, but is required by 10 CFR 50.55. DOC A.21 is revised. DOC LA.11 is deleted and DOC L.36 is added to discuss the relocation of the steam generator support inspections to the Inservice Inspection Program.

The ITS and Bases are unaffected.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

(A.22)

5.5.6 4.4.10.1.1 In addition to the requirements of Specification 4.0.5, the Reactor Coolant pump flywheels shall be inspected once every 10 years by a qualified inplace UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces defined by the volume of disassembled flywheels.

5.5.7 ~~4.4.10.1.2 In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval.~~ (L.36) R13

A.1

ITS 5.0

04-22-98

ITS

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 & 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3/4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

See CTS 3.4.10.1

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

5.5.6 4.4.10.1.1 ~~In addition to the requirements of Specification 4.0.5,~~ the Reactor Coolant pump flywheels shall be inspected once every 10 years by a qualified inplace UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces defined by the volume of disassembled flywheels. A.22

5.5.7 ~~4.4.10.1.2 In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval.~~ L.36 | R13

inspection requirements for ASME Code Class 1, 2, and 3 components. ITS 5.5.7 does not include the statement in CTS 4.0.5.a and does not include references to inservice inspection. This changes the CTS by not including a reference to 10 CFR 50.55a requirements or references to ASME Code Class 1, 2, and 3 inservice inspection. The 10 CFR 50.55a requirements are still applicable without the reference. R13

This change is acceptable because the 10 CFR Part 50 requirements are still applicable and referencing them separately is unnecessary. This change is designated administrative because it does not result in technical changes to the CTS. R13

A.22 CTS 4.4.10.1.1 states, "In addition to the requirements of Specification 4.0.5, the Reactor Coolant pump flywheels shall be inspected..." ITS 5.5.6 does not include the reference to Specification 4.0.5, which is ITS 5.5.7, Inservice Testing Program. This changes the CTS by not referencing CTS 4.0.5 requirements which are required regardless of the reference.

This change is acceptable because it deletes a reference to a requirement that has it's own criteria for application, regardless of the reference. This change is designated administrative because it does not result in technical changes to the CTS.

A.23 ITS 5.5.10 states, "The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies." CTS 4.7.7 and CTS 4.7.8 do not explicitly state these allowances, but they apply as CTS 4.0.2 and CTS 4.0.3, which are equal to ITS SR 3.0.2 and SR 3.0.3, because these allowances apply to all the CTS LCO Surveillance Requirements. This changes the CTS by explicitly invoking the allowances of ITS SR 3.0.2 and ITS SR 3.0.3 because the requirements have been moved to Section 5.0, and an explicit allowance is needed to retain the existing allowances.

This change is acceptable because it retains existing allowances by transferring them into ITS format. This change is designated administrative because it does not result in technical changes to the CTS.

A.24 CTS 6.9.1 states, "In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted." ITS 5.6 states, "The following reports shall be submitted in accordance with 10 CFR 50.4." This changes the CTS by referencing 10 CFR 50.4 as the reference for how to submit reports and excluding the remaining detail, which is already addressed in 10 CFR 50.4.

require such a program. This changes the CTS by moving the requirements for the Radiological Environmental Monitoring Program to the ODCM.

The purpose of CTS 6.8.4.f is to provide representative measurements of radioactivity in the highest potential exposure pathways, and verification of the accuracy of the effluent monitoring program. The removal of the requirement for this program 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. ITS 5.6.2 still requires an annual report of the results of the "Radiological Environmental Monitoring Program." Also, this change is acceptable because these types of procedural details will be adequately controlled in the ODCM. This change is designated as a less restrictive, removal of detail, because the requirements for a program are being removed from the Technical Specifications.

LA.11 Not used.

R13

LA.12 *(Type 3 – Removing Procedural Details for Meeting TS Requirements and Related Reporting Problems)* CTS 4.6.1.1.c states, "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." ITS 5.0 does not include such a specific requirement for the equipment hatch. This changes the CTS by moving the reference leak rate testing for the equipment hatch to the Containment Leak Rate Testing Program (CLRTP).

RAI 3.6.1-5 R1

The removal of these details for performing surveillance requirements from the Technical Specifications is acceptable because this type of information is not

changes the CTS by changing the description of the frequency for the integrated leak test requirements to 18 months, and allowing the test to be performed within 1.25 times the 18 month interval. This interval could be longer or shorter than the "refueling interval" frequency.

The purpose of CTS 6.8.4.a(ii) is to assure that the integrated leak test requirements are met at least every refueling interval. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The change still assures the integrated leak test requirements are met at least every refueling interval, but the description of the frequency is changed to be consistent with similar requirements in the ISTS. This change is designated as less restrictive because Surveillance could be performed less frequently under the ITS than under the CTS. R4

L.36 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.4.10.1.2 states, "In addition to the requirements of Specification 4.0.5, at least one third of the main member to main member welds, joining A572 material, in the steam generator supports, shall be visually examined during each 40 month inspection interval." The ITS does not contain this requirement. This changes the CTS by eliminating the Technical Specifications requirement for visual inspection of the steam generator supports. R13

This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the values used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Inspection of plant components will continued to be performed, as appropriate, under the Inservice Inspection Program required by 10 CFR 50.55. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

Specifications Affected: ITS 5.0 Determination of No Significant Hazards Consideration

Description

In Supplement 4 to the North Anna ITS license amendment, Section 5.0, page 38 of the Determination of No Significant Hazards Consideration section of the submittal was affected by changes made to other pages, but was not included in the supplement. The page is provided in this supplement.

The ITS and Bases are unaffected.

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

   This change eliminates the requirement to project dose contributions for radioactive effluents for the current calendar quarter and the current calendar year. The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change will not introduce new accident initiators. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change eliminates the requirement to project dose contributions for radioactive effluents for the current calendar quarter and the current calendar year. The ITS requirements are considered to provide adequate monitoring of dose contributions from radioactive effluents. As a result, the change does not significantly reduce the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES

SPECIFICATION 5.0, CHANGE L.32

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." The proposed change involves making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the determination of No Significant Hazards Considerations for conversion to NUREG-1431.

L.32 CTS 1.22 describes the Process Control Program (PCP). CTS 6.14 (Unit 1) and CTS 6.13 (Unit 2) specifies the change control for the PCP. CTS 6.8.1.g requires written procedures be established, implemented, and maintained to cover PCP implementation. The ITS does not specify requirements for the PCP. This changes the CTS by removing the requirements associated with the contents and maintenance of the PCP.

The purpose of CTS 1.22, CTS 6.14 (Unit 1), CTS 6.13 (Unit 2), and 6.8.1.g is to describe requirements for the PCP in order to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste. This change is acceptable because the requirements for the PCP change control are not required to be in the ITS to provide adequate protection of the public health and safety. The requirements of 10 CFR Parts 20, 61, and 71 will continue to be complied with, and NAPS will also continue

Summary of Changes to the NAPS ITS Submittal
Miscellaneous Changes

Specifications Affected: ITS 5.5.7, 3.4.10, 3.4.11, 3.4.14, 3.5.2, 3.6.6, 3.6.7, 3.7.1, 3.7.2, 3.7.5, and 3.8.1.

Description

ITS 5.5.7, "Inservice Testing Program," and the Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. This affects ITS 5.5.7, ISTS 5.5.7 markup, adds ITS JFD 24, the Unit 1 and Unit 2 CTS markup, adds DOC A.39, and the ITS Bases, ISTS Bases markup, and ITS Bases JFD for 3.4.10, 3.4.11, 3.4.14, 3.5.2, 3.6.6, 3.6.7, 3.7.1, 3.7.2, 3.7.5, and 3.8.1.

5.5 Programs and Manuals

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in the ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda as follows: R13

ASME Code for Operation and Maintenance of Nuclear Power Plants and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities R13
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Code for Operation and Maintenance of Nuclear Power Plants shall be construed to supersede the requirements of any TS. R13

5.5.8 Steam Generator (SG) Tube Surveillance Program

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test Frequencies.

CTS

5.5 Programs and Manuals (continued)

4.0.5
4.4.10.1.2

5.5.~~8~~ 7 **Inservice Testing Program** 7

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following: TSTF-279

4.0.5.b

a. Testing frequencies specified in ~~Section XI of~~ the ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda as follows: 24 | R13

for Operation and Maintenance of Nuclear Power Plants 24 | R0

ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME ~~Boiler and Pressure Vessel~~ Code shall be construed to supersede the requirements of any TS. 24 | R13

4.4.5.2

5.5.~~9~~ 8 **Steam Generator (SG) Tube Surveillance Program** 7

INSERT

~~[Reviewer's Note: The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.]~~ 1

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test Frequencies. TSTF-118

(continued)

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changes are consistent with the current licensing basis and guidance in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," section 4.4.

19. ISTS 5.5.6 is modified to state that the provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program surveillance frequency. This allowance is consistent with the current licensing basis, and is consistent with the NUREG-1431 format of retaining these allowances for other current Technical Specification requirements that have been moved to Section 5.0.

20. The discussion in ISTS 5.3.1 regarding qualifications of staff not covered by Regulatory Guide 1.8 is replaced with a statement that the shift supervisor, assistant shift supervisor, Control Room Operator – Nuclear, and the individual providing advisory support to the unit operations shift crew are required to meet or exceed the minimum qualifications of 10 CFR 55.59(c) and 55.31(a)(4). These requirements are consistent with the CTS, and the CTS requirements do not include qualifications of staff not covered by Regulatory Guide 1.8. RAI 5.0-01 R4

21. References in ISTS 5.5.12 to the "offgas system" are not adopted. NAPS does not include an offgas system, which is usually associated with boiling water reactors. RAI 5.0-07 R4

22. This bracketed requirement is deleted because it is not applicable to North Anna. The following requirements are renumbered, where applicable, to reflect this deletion. RAI 5.0-08 R4

23. The requirement to include a preplanned alternate method of monitoring in case of Post Accident Monitoring (PAM) instrumentation inoperability is not adopted. The NAPS design does not have alternate methods of monitoring if the PAM instrumentation is inoperable. R4

24. ITS 5.5.7 is revised to refer to the "ASME Code for Operation and Maintenance of Nuclear Power Plants" instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program. R13

A.1

8-5-80

ITS

5.5 APPLICABILITY

5.5.7 SURVEILLANCE REQUIREMENTS (Continued)

A.39 | R13

A.21

5.5.7.a b. Surveillance intervals specified in ~~Section XI of~~ the ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda for the inservice ~~inspection and~~ testing activities required by the ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda shall be applicable as ~~follows in these~~ Technical Specifications: A.39 | R13

for the Operation and Maintenance of Nuclear Power Plants

ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

13

A.13

5.5.7.b c. The provisions of Specification ~~4.0.2~~ SR3.0.2 are applicable to the above required frequencies for performing inservice ~~inspection and~~ testing activities. A.21 A.20

13

d. ~~Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.32

5.5.7.d e Nothing in the ASME ~~Boiler and Pressure Vessel~~ Code shall be construed to supersede the requirements of any Technical Specification. A.39 | R13

L.20

Insert proposed ITS 5.5.7.c

A.1

ITS 5.0

8-21-80

ITS

APPLICABILITY

5.5.7. SURVEILLANCE REQUIREMENTS (Continued)

5.5.7a b. ~~Surveillance intervals~~ specified in ~~Section XI of~~ the ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda for the inservice ~~inspection and~~ testing activities required by the ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda shall be applicable as ~~follows in these Technical Specifications:~~ A.39 A.21 R13

for the Operation and Maintenance of Nuclear Power Plants

ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

A.39 R13

A.13

5.5.7.b c. The provisions of Specification ~~4.0.2~~ SR 3.0.2 are applicable to the above required frequencies for performing inservice ~~inspection and~~ testing activities. A.21 A.20

d. ~~Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.~~ A.32

5.5.7.d e. Nothing in the ASME ~~Boiler and Pressure Vessel~~ Code shall be construed to supersede the requirements of any Technical Specification. A.39 R13

Insert proposed ITS 5.5.7.c

L.20

This change is acceptable because it clarifies the intent of the new program, which incorporates existing requirements into a separate program without changing the requirements. This change is designated administrative because it does not result in technical changes to the CTS. R4

A.39 CTS 4.0.5.b refers to Section XI and ASME, Boiler and Pressure Vessel Code." ITS 5.5.7 refers to the "ASME Code for Operation and Maintenance of Nuclear Power Plants" and does not reference Section XI. This changes the CTS by revising the title of the applicable ASME Code for the Inservice Testing Program to match the currently approved version.

This change is acceptable because North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). This version of the Code does not use Section XI for the Inservice Testing Program and is called the "ASME Code for Operation and Maintenance of Nuclear Power Plants" instead of the "ASME, Boiler and Pressure Vessel Code." This change is designated administrative because it does not result in technical changes to the CTS. R13

MORE RESTRICTIVE CHANGES

M.1 ITS 5.1.1 states, "The plant manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety." The CTS does not include such a statement. This changes the CTS by adding a required action for the plant manager or his designee.

The purpose of the ITS 5.1.1 statement is to provide additional assurance that the plant manager has direct responsibility for overall unit operation. This change is acceptable because having the plant manager or his designee approve actions affecting nuclear safety is consistent with the ITS 5.2.1.b requirement, "The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant." This change is designated more restrictive because an additional requirement is added to the Technical Specifications.

M.2 ITS 5.4.1 states, "Written procedures shall be established, implemented, and maintained covering the following activities:…b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33." The CTS does not include this requirement. This changes the CTS by adopting a new requirement for emergency operating procedures.

The purpose of ITS 5.4.1.b is to ensure that written procedures are established, implemented, and maintained covering the emergency operating procedures to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ 235°F (Unit 1), 270°F (Unit 2) within 24 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. With any RCS cold leg temperatures at or below 235°F (Unit 1), 270°F (Unit 2), overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified. R13

The pressurizer safety valve setpoint given in the LCO is for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Chapter 15.
3. WCAP-7769, Rev. 1, June 1972.

BASES

REFERENCES (continued)	4. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13

BASES

ACTIONS

A.1 (continued)

coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

unit

If the Required Action of A.1 cannot be met within the required Completion Time or if two ~~or more~~ pressurizer safety valves are inoperable, the ~~plant~~ must be brought to a MODE in which the requirement does not apply. To achieve this status, the ~~plant~~ must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ [275]°F within ~~12~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~plant~~ conditions from full power conditions in an orderly manner and without challenging ~~plant~~ systems. With any RCS cold leg temperatures at or below [275]°F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by [three] pressurizer safety valves.

Unit

24

235°F (Unit 1), 270°F (Unit 2)

Unit

① ① TSTF-352 ② ① ① ② ②

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of ~~Section XI of~~ the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

⑧ 1R13

The pressurizer safety valve setpoint is ~~± [3]%~~ given in the LCO for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

③

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. U FSAR, Chapter [15].

② ①

(continued)

Rev 13

BASES

REFERENCES (continued)

3. WCAP-7769, Rev. 1, June 1972.

4. ASME, ~~Boiler and Pressure Vessel Code, Section XI.~~ 8 | R13

ASME Code for Operation and Maintenance of Nuclear Power Plants.

Rev.0

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. North Anna has been approved to use a pressurizer safety valve tolerance of +2% / -3% average as-found with no single valve outside of ±3%. The Bases have been revised accordingly.

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. The list of events considered in the overpressure protection analysis was revised to be consistent with the most recent analysis.

6. The Bases description of the Applicability Note, which allows pressurizer safety valves to be tested at pressure and temperature, is modified. North Anna does not currently utilize that testing method (the pressurizer safety valves are removed and bench tested), but the testing method is an acceptable approach and the option to use this method was adopted. The Bases are modified to state that the allowed time for the testing is based on industry experience instead of plant operating experience.

7. The Bases are revised to match the Specifications.

8. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

ACTIONS

H.1 and H.2 (continued)

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, automatic PORV OPERABILITY is required. See LCO 3.4.12.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

SR 3.4.11.1 requires verification that the pressure in the PORV backup nitrogen system is sufficient to provide motive force for the PORVs to cope with a steam generator tube rupture coincident with loss of the containment Instrument Air system. The Frequency of 7 days is based on operating experience.

SR 3.4.11.2

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code (Ref. 3). R13

This SR is modified by two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed, in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.

Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

SR 3.4.11.3

SR 3.4.11.3 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. This testing is performed in MODES 3 or 4 to prevent possible RCS pressure transients with the reactor critical. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.3 (continued)

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

SR 3.4.11.4

Operating the solenoid control valves and check valves on the accumulators ensures the PORV control system actuates properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. UFSAR, Section 15.4.
3. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13

Insert 1

BASES (continued)

8

SURVEILLANCE REQUIREMENTS

SR 3.4.11.~~1~~ 2

8

opened and

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 3). ~~If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR).~~ 10 | R13

~~that is incapable of being manually cycled~~

TSTF-151

TSTF-284

This SR is modified by two Notes.

~~The~~ Note 2 modifies this SR by stating that it is not required to be ~~met~~ performed with the block valve closed, in accordance with the Required Action 5 of this LCO. TSTF-284

Insert 2

6

SR 3.4.11.~~2~~ 3 8

SR 3.4.11 ~~2~~ requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice. 5 TSTF-284

This testing is performed in MODES 3 or 4 to prevent possible RCS pressure transients with the reactor critical.

Insert 3

4

~~SR 3.4.11.3~~ 8

~~Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of [18] months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.~~ 5 5

~~SR 3.4.11.4~~

~~This Surveillance is not required for plants with permanent 1E power supplies to the valves.~~ 3

(continued)

Rev 13

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.4.11.4 (continued)~~

~~The Surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

3

REFERENCES

1. Regulatory Guide 1.32, February 1977. 1 5
2. U FSAR, Section ~~[15.2]~~ 15.4.
3. ASME ~~, Boiler and Pressure Vessel Code, Section XI.~~ 10 | RB

ASME Code of Operation and Maintenance of Nuclear Power Plants.

Rev 13

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

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

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

6. Information on the testing of each PORV through a complete cycle is relocated from the CTS to the Bases for ITS SR 3.4.11.3. A conflicting statement is removed from the ACTION Note Bases.

7. The North Anna PORV design includes one PORV which opens at a specific pressure and a second PORV that opens at a set fraction of an operator-controlled setpoint. The Bases have been changed to reflect this design.

8. The Bases are revised to reflect the North Anna analysis assumptions regarding the PORV backup nitrogen supply. The PORVs are air operated valves and are supplied with a backup nitrogen supply which can provide motive force to the PORVs should the containment Instrument Air system fail. An ACTION is added for inoperability of the PORVs due to failure of the backup nitrogen system. A Surveillance is added to verify sufficient pressure in the nitrogen backup system. These changes are consistent with the CTS and the North Anna accident analysis.

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

10. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section R13

XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. 6), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. R13

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures. If testing cannot be performed at these pressures, testing can be performed at lower pressures and scaled to operating pressure.

Entry into MODES 3 and 4 is allowed if needed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on any RCS PIVs in the RHR System flow path when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path that are required to be tested must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. UFSAR, Section 3.1.48.1.

BASES

REFERENCES (continued)

4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. Letter from D. G. Eisenhut, NRC, to all LWR licensees, LWR Primary Coolant System Pressure Isolation Valves, February 23, 1980.
6. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13
7. 10 CFR 50.55a(g).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

7

10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, ~~Section XI~~ (Ref. 9), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

6

7

9 7 IR13

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

If testing cannot be performed at these pressures, testing can be performed at lower pressures and scaled to operating pressures.

Entry into MODES 3 and 4 is allowed if needed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on any RCS PIVs in the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path that are required to be tested must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

8

7

~~SR 3.4.14.2 and SR 3.4.14.3~~

~~Verifying that the RHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of [600] psig. The interlock setpoint that prevents the valves from being~~

5

(continued)

Rev. 13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.2 and SR 3.4.14.3 (continued)

opened is set so the actual RCS pressure must be ≤ [425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The [18] month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The [18] month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.

5

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. ~~10 CFR 50, Appendix A, Section V, GDC 55~~ UFSAR, Section 3.1.48.1. 1
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. ~~NUREG-0677, May 1980.~~ 7
6. ~~[Document containing list of PIVs.]~~ 9 R13
6 7. ASME Boiler and Pressure Vessel Code, Section XI. ASME Code for Operation and Maintenance of Nuclear Power Plants.
7 8. 10 CFR 50.55a(g).

Letter from D. G. Eisenhut, NRC to all LWR licensees, LWR Primary Coolant System Pressure Isolation Valves, Feb. 23, 1980.

Insert 1

Insert 2 7

Rev 13

1. North Anna Units 1 and 2 were designed and constructed on the basis of the proposed General Design Criteria, published in 1966. Since February 20, 1971, when the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50, were published, the Company attempted to comply with the intent of the newer criteria to the extent practical, recognizing previous design commitments. The NRC's Safety Evaluation Report for North Anna Units 1 and 2 reviewed the plant against 10 CFR Part 50, Appendix A and concluded that the facility design conforms to the intent of the newer criteria. The North Anna UFSAR contains discussions comparing the design of the plant to the 10 CFR 50, Appendix A, General Design Criteria. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

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

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

6. Details of Surveillance testing are relocated from the CTS to the Bases.

7. The Bases have been modified to reflect the North Anna licensing basis and the list of PIVs that must be tested. Based on an NRC Order and the Unit 2 SER, only a subset of PIVs are required to be leak tested under the Technical Specifications. Those PIVs, which were determined by the NRC to match the WASH-1400 configurations that could cause an intersystem LOCA bypassing containment, are tested to the Specification's requirements. The list of valves is located in the ITS Bases. Unused references are eliminated and subsequent references are renumbered.

8. The Surveillance Bases are modified to clarify that the PIV testing can be performed at lower differential pressures and scaled to operating pressure.

9. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section R13

XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3 (continued)

void size are gradual in nature, and the system is operated in accordance with procedures to preclude growth in these voids.

To provide additional assurances that the system will function, a verification is performed every 92 days that the system is sufficiently full of water. The system is sufficiently full of water when the voids and pockets of entrained gases in the ECCS piping are small enough in size and number so as to not interfere with the proper operation of the ECCS. Verification that the ECCS piping is sufficiently full of water can be performed by venting the necessary high point ECCS vents outside containment, using NDE, or using other Engineering-justified means. Maintaining the piping from the ECCS pumps to the RCS sufficiently full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of excess noncondensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 92 day frequency takes into consideration the gradual nature of the postulated void generation mechanism.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. R13
This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the safety analysis. SRs are specified in the Inservice Testing Program, which encompasses the ASME Code. R13
The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump capable of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.2 (continued)

under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

some sufficiently

With the exception of the operating ~~centrifugal~~ charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of excess noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. ~~The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.~~ INSERT 17 19 92 17

of the postulated void generation mechanism

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~Section XI of~~ the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the ~~plant~~ safety analysis. SRs are specified in the Inservice Testing Program, which encompasses ~~Section XI of~~ the ASME Code. ~~Section XI of the~~ ASME Code provides the activities and Frequencies necessary to satisfy the requirements. 20 | R13 2 20 | R13

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or

(continued)

Rev. 13

18. North Anna Units 1 and 2 were designed and constructed on the basis of the proposed General Design Criteria, published in 1966. Since February 20, 1971, when the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50, were published, the Company attempted to comply with the intent of the newer criteria to the extent practical, recognizing previous design commitments. The NRC's Safety Evaluation Report for North Anna Units 1 and 2 reviewed the plant against 10 CFR Part 50, Appendix A and concluded that the facility design conforms to the intent of the newer criteria. The North Anna UFSAR contains discussions comparing the design of the plant to the 10 CFR 50, Appendix A, General Design Criteria. Bases references to the 10 CFR 50, Appendix A criteria have been replaced with references to the appropriate section of the UFSAR.

19. A Frequency of 92 days is adopted for SR 3.5.2.3 to verify that ECCS piping is sufficiently full of water. The 92 day Frequency has been determined to be adequate based on plant operating experience and engineering analysis. Performing the SR every 92 days does not verify the ECCS piping completely filled with water, but provides an added degree of assurance that the piping is sufficiently full of water to allow the ECCS to perform its function when required. There is no requirement for this Surveillance in the CTS.

20. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1 (continued)

since they were verified to be in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

Verifying that each QS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that QS pump performance is consistent with the safety analysis assumptions. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 4). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program. R13

SR 3.6.6.3 and SR 3.6.6.4

These SRs ensure that each QS automatic valve actuates to its correct position and each QS pump starts upon receipt of an actual or simulated Containment Pressure high-high 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 these Surveillances under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at an 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.5

With the quench spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.5 (continued)

spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzle and the non-corrosive design of the system, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the QS System provides assurance that the proper flow path exists for QS System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

is consistent with the safety analysis assumptions 3

SR 3.6.6.2 1

Verifying that each QS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that QS pump performance ~~has not degraded during the cycle~~. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref. 4). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on ~~bypass~~ flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

8 / RB

recirculation 4

SR 3.6.6.3 and SR 3.6.6.4 1

These SRs ensure that each QS automatic valve actuates to its correct position and each QS pump starts upon receipt of an actual or simulated containment ~~spray actuation~~ signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform these Surveillances under the conditions that apply during a ~~plant~~ outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually

Pressure-high-high 4

5

unit 3

(continued)

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1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6D.3 and SR 3.6.6D.4 (continued)

1

5

pass the Surveillances when performed at an [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

and the non-corrosive design of the system

3 1

SR 3.6.6D.5

quench

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzle, a test at ~~[the first refueling and at]~~ 10 year intervals is considered adequate to detect obstruction of the nozzles.

2

3 5

REFERENCES

1. UFSAR, Section [6.2].
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

8 /R13

ASME Code for Operation and Maintenance of Nuclear Power Plants.

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1. North Anna Units 1 and 2 utilize subatmospheric containments. Therefore, the NUREG-1431 specifications applicable to that containment design were used in developing the plant-specific Improved Technical Specifications (ITS). Necessary editorial changes to the NUREG-1431 pages were made.

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. 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.

4. Information is moved from the current Technical Specifications to the Bases.

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

6. 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.

7. The ITS Bases ASA section adds the phrase "during a DBA" to qualify that the parameters being discussed, Service Water temperature, RWST water temperature, and the containment air temperature affect containment pressure during a DBA and not normal operation as the ISTS paragraph could imply. This deviation is acceptable because it reflects the NAPS analysis.

8. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.7.5 (continued)

differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 4). Since the RS System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program. R13

SR 3.6.7.6

These SRs ensure that each automatic valve actuates and that the RS System and casing cooling pumps start upon receipt of an actual or simulated High-High containment pressure signal. Start delay times are also verified for the RS System pumps. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was considered to be acceptable from a reliability standpoint.

SR 3.6.7.7

This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment will meet its design bases objective. An air or smoke test is performed through each spray header. Due to the passive design of the spray header and its normally dry state, a test at 10 year intervals is considered adequate for detecting obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 6.2.
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.

BASES

REFERENCES (continued)	4. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.~~6E~~ 7.4 (continued) 1

otherwise secured in position, since they are verified as being in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.~~6E~~ 7.5 1

Verifying that each RS ~~[and casing cooling]~~ pump's developed head at the flow test point is greater than or equal to the required developed head ensures that these pumps' performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref. 4). Since the ~~QS~~ RS System pumps cannot be tested with flow through the spray headers, they are tested on ~~bypass~~ recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program. 2 13 IR13 2 3

SR 3.6.~~6E~~ 7.6 1

These SRs ensure that each automatic valve actuates and that the RS System and casing cooling pumps start upon receipt of an actual or simulated High-High containment pressure signal. Start delay times are also verified for the RS System pumps. 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 the Surveillance when performed at the ~~[18]~~ month Frequency. Therefore, the Frequency was considered to be acceptable from a reliability standpoint. 4 3 4

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.~~6E~~.7 7 1

This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment will meet its design bases objective. An air or smoke test is performed through each spray header. Due to the passive design of the spray header and its normally dry state, a test at ~~[the first refueling and at]~~ 10 year intervals is considered adequate for detecting obstruction of the nozzles. 7

REFERENCES

1. ~~U~~FSAR, Section ~~[~~6.2~~]~~. 2
2. 10 CFR 50.49.
3. 10 CFR 50, Appendix K.
4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ 13 | R13

ASME Code for Operation and Maintenance of Nuclear Power Plants.

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generator. References to the worst case single active failure are modified to reflect this plant specific assumption.

10. In ISTS 3.6.6E Background and Action A.1 Bases sections, the word "approximately" is added to "50%" and "150%," respectively. This is in reference to the RS heat removal capability of one RS subsystem and 3 RS subsystems, respectively. The exact capacity for each RS subsystem varies, but is approximately 50%. Adding this change makes the statements more accurate.

11. Information is added to the Bases for ITS Required Action D.1 to clarify available RS cooling capability when the casing cooling tank is inoperable.

12. Information is added to the LCO Bases to clearly define what is required for an OPERABLE RS subsystem in accordance with the ITS Writer's Guide. R13

13. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

ACTIONS

B.1 and B.2 (continued)

Action B.2 to reduce the setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Protection System Instrumentation," provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code (Ref. 4) R13

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

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

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

REFERENCES

1. UFSAR, Section 10.3.1.

2. ASME, Boiler and Pressure Vessel Code, Section III, 1968 Edition with Addenda through Winter 1970.

3. UFSAR, Section 15.2.

4. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13

5. ANSI/ASME OM-1-1987.

BASES

ACTIONS (continued)

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

Required Actions are not completed

TSTF-235

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

≥4 inoperable

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code ~~, Section XI~~ (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required: (6) | R13

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

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

Insert paragraph from Page B 3.7-2

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

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

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

REFERENCES

1. U FSAR, Section ~~[~~10.3.1~~]~~. ① ②
2. ASME, Boiler and Pressure Vessel Code, Section III, ~~Article NC-7000, Class 2 Components~~. 1968 Edition with Addenda through Winter 1970 ①
3. U FSAR, Section ~~[~~15.2~~]~~. ① ②
4. ASME, ~~Boiler~~ and ~~Pressure~~ Vessel Code~~, Section~~ XI. ⑥ IR13
5. ANSI/ASME OM-1-1987.
6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994. TSTF-235

ASME Code for Operation and Maintenance of Nuclear Power Plants.

Rev. 13

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. Changes are made to reflect changes made to the specifications.

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

6. The Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

ACTIONS

C.1 and C.2 (continued)

For inoperable MSTVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSTVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSTV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSTVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSTV isolation time is ≤ 5.0 seconds. The MSTV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSTVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSTVs are not tested at power, they are exempt from the ASME Code (Ref. 5) requirements during operation in MODE 1 or 2. R13

The Frequency is in accordance with the Inservice Testing Program.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies that each MSTV closes on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MSTV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 6.2.
3. UFSAR, Section 15.4.2.
4. 10 CFR 100.11.
5. ASME Code for Operation and Maintenance of Nuclear Power Plants. R13

BASES

ACTIONS

C.1 and C.2 (continued)

The ~~[8]~~ 8 hour Completion Time is consistent with that allowed in Condition A. ②

For inoperable MSIVs MSTVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs MSTVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV MSTV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position. ①

D.1 and D.2 ①

If the MSIVs MSTVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 ① ②

This SR verifies that MSIV MSTV closure isolation time is ≤ ~~[4.6]~~ 5.0 seconds ~~on an actual or simulated actuation signal~~. The MSIV MSTV closure isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs MSTVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs MSTVs are not tested at power, they are exempt from the ASME Code, ~~Section XI~~ (Ref. 5), requirements during operation in MODE 1 or 2. TSTF-~~289~~ ① ① ④ | R13

(continued)

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 ①

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

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

TSTF-289 ①

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

Insert

REFERENCES

1. UFSAR, Section [10.3]. ① ②
2. UFSAR, Section [6.2]. ① ②
3. UFSAR, Section ~~[15.1.5]~~ 15.4.2 ① ②
4. 10 CFR 100.11.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI~~ ASME Code for Operation and Maintenance of Nuclear Power Plants. ④ | R13

Rev. 13

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 Bases are revised to refer to the ASME Code and reference the "ASME Code for Operation and Maintenance of Nuclear Power Plants" when discussing the Inservice Testing Program, instead of referencing Section XI of the ASME Code and "ASME, Boiler and Pressure Vessel Code, Section XI." North Anna has adopted the ASME Code for Operation and Maintenance of Nuclear Power Plants, the 1995 Edition and the 1996 Addenda, as required by 10 CFR 50.55a(b)(3). With this adoption, references to Section XI and to the ASME Boiler and Pressure Vessel Code are incorrect when discussing the Inservice Testing Program in the North Anna ITS and Bases. R13

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 2). Because it is sometimes undesirable to introduce cold AFW into the steam generators while they are operating, this testing is typically performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement. R13

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

SR 3.7.5.3

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

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually align the required valves.