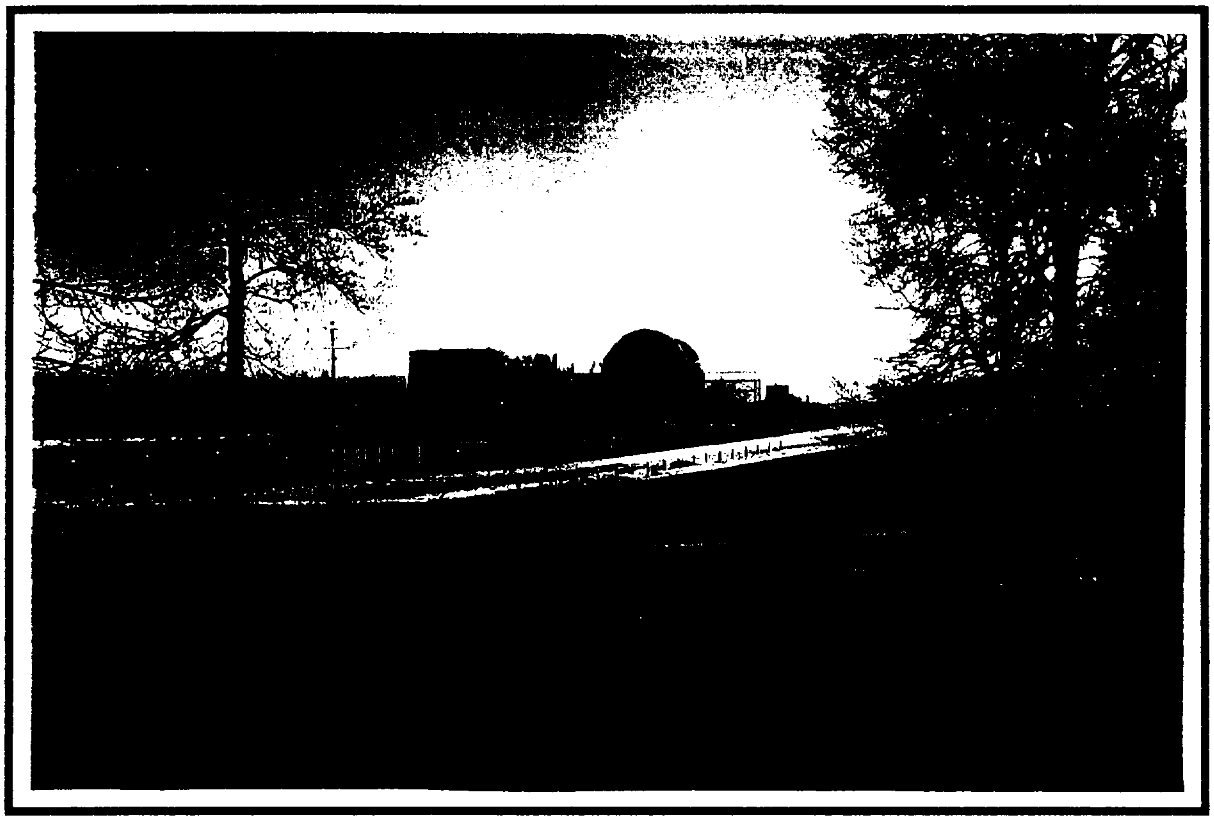
NORTH ANNA POWER STATION

Section 3.9
Refueling Operations



VOLUME 19
Improved Technical Specifications

Dominion

NORTH ANNA POWER STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.9 - REFUELING OPERATIONS

SECTION 3.9 - REFUELING OPERATIONS

IMPROVED TECHNICAL SPECIFICATIONS

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

------------------------------NOTE------------------------------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within the limit specified in COLR.	72 hours

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3.9 REFUELING OPERATIONS

3.9.2 Primary Grade Water Flow Path Isolation Valves—MODE 6

LCO 3.9.2 Each valve used to isolate primary grade water flow paths shall be secured in the closed position.

-------------------- NOTE --------------------
Primary grade water flow path isolation valves may be opened under administrative control for planned boron dilution or makeup activities.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more valves not secured in closed position.	A.1	Suspend positive reactivity additions.	Immediately
	<u>AND</u>		
	A.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	A.3	Secure valves in closed position.	15 minutes
	<u>AND</u>		
	A.4	Perform SR 3.9.1.1.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Verify each valve in the affected flow path that isolates primary grade water flow paths is locked, sealed, or otherwise secured in the closed position.	Within 15 minutes following a boron dilution or makeup activity

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One source range neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
B. Two source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND	
	B.2 Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.3.2	-------------------NOTE-------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in each installed air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

   2. capable of being closed by an OPERABLE containment purge and exhaust isolation valve.

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

1. Not applicable to the 7 ft containment personnel air lock.

2. Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2	Verify each required containment purge and exhaust valve actuates to the isolation position on manual initiation.	18 months

3.9 REFUELING OPERATIONS

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

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

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

The required RHR loop may not be in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System (RCS), coolant of boron concentration less than required to meet the minimum required boron concentration of LCO 3.9.1.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Close equipment hatch and secure with four bolts.	4 hours
	AND	
	A.5 Close one door in each installed air lock.	4 hours
	AND	
	A.6.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR	
	A.6.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm.	12 hours

3.9 REFUELING OPERATIONS

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

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

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

1. All RHR pumps may be de-energized for ≤ 15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10°F below saturation temperature;

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration; and

   c. No draining operations to further reduce RCS volume are permitted.

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other loop is OPERABLE and in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	<u>OR</u>	
	A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No RHR loop in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
	B.2 Initiate action to restore one RHR loop to operation.	Immediately
	<u>AND</u>	
	B.3 Close equipment hatch and secure with four bolts.	4 hours
	<u>AND</u>	
	B.4 Close one door in each installed air lock.	4 hours
	<u>AND</u>	
	B.5.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	<u>OR</u>	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.5.2 Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of: a. ≥ 3000 gpm, or b. ≥ 2000 gpm if RCS temperature ≤ 140°F and time since entry into MODE 3 ≥ 100 hours.	12 hours
SR 3.9.6.2	-------------------NOTE-------------------- Not required to be performed until 24 hours after a required RHR pump is not in operation. --- Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days

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3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.7.1 Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

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SECTION 3.9 - REFUELING OPERATIONS

IMPROVED TECHNICAL SPECIFICATIONS BASES

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the Refueling Water Storage Tank through the open reactor vessel by gravity feeding or by the use of the Low Head Safety Injection System pumps.

The pumping action of the Residual Heat Removal (RHR) System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during

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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 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. 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

APPLICABILITY (continued)

LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

The applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and refueling cavity when those volumes are connected to the RCS. When the refueling canal and refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

A.3

In addition to immediately suspending CORE ALTERATIONS and positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

(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 in each required volume 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.

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.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Primary Grade Water Flow Path Isolation Valves—MODE 6

BASES

BACKGROUND

During MODE 6 operations, the isolation valves for primary grade water flow paths that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be locked, sealed or otherwise secured in the closed position.

The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by uncontrolled reduction of the boron concentration is inappropriate during MODE 6, isolation of all primary grade water flow paths prevents an unplanned boron dilution.

APPLICABLE SAFETY ANALYSES

The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that primary grade water flow paths be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate primary grade water flow paths. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating primary grade water flow paths, a safety analysis for an uncontrolled boron dilution accident is not required for MODE 6.

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

LCO

This LCO requires that flow paths to the RCS from primary grade water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.

For Unit 1, primary grade water flow paths may be isolated from the RCS by closing valve 1-CH-217 or 1-CH-220, 1-CH-241, FCV-1114B and FCV-1113B. For Unit 2, primary grade water flow paths may be isolated from the RCS by closing valve 2-CH-140, or 2-CH-160, 2-CH-156, FCV-2114B, and FCV-2113B.

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BASES

LCO (continued)	The LCO is modified by a Note which allows the primary grade water flow path isolation valves to be opened under administrative control for planned boron dilution or makeup activities.
APPLICABILITY	In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of primary grade water flow paths to the RCS. In MODES 3, 4, and 5, LCO 3.1.8, Primary Grade Water Flow Path Isolation Valves, requires the primary grade water flow paths to the RCS to be isolated to prevent an inadvertent boron dilution. In MODES 1 and 2, the boron dilution accident was analyzed and was found to be capable of being mitigated.
ACTIONS	A.1 Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the primary grade water flow path isolation valves locked, sealed, or otherwise secured closed, except as allowed under administrative control by the LCO Note. Because of the possibility of an inadvertent boron dilution, other positive reactivity additions and CORE ALTERATIONS must be prohibited while securing the isolation valves on the unborated water systems. The Completion Time of "Immediately" for suspending positive reactivity additions and CORE ALTERATIONS reflects the importance of preventing known positive reactivity additions so that any boron dilution event can be readily identified and terminated. A.2 Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. With any valve used to isolate primary grade water flow paths not locked, sealed or otherwise secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

BASES

ACTIONS (continued)

A.3

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the primary grade water flow path isolation valves secured closed. Locking, sealing, or securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of 15 minutes provides sufficient time to close, lock, seal, or otherwise secure the flow path isolation valve.

A.4

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed to demonstrate that the required boron concentration exists. The Completion Time of 1 hour is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

These valves are to be locked, sealed, or otherwise secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that the primary grade water flow paths are isolated, precluding a dilution. The boron concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. The Frequency is based on verifying that the isolation valves are locked, sealed, or otherwise secured within 15 minutes following a boron dilution or makeup activity. This Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. UFSAR, Section 15.2.4.

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B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps). The detectors also provide continuous visual indication and an audible alarm in the control room to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Primary Grade Water Flow Path Isolation Valves—MODE 6."

The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

BASES

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1 must be suspended immediately. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operations. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

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BASES

ACTIONS	B.2 (continued) The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.
SURVEILLANCE REQUIREMENTS	SR 3.9.3.1 SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions. The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1. SR 3.9.3.2 SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.
REFERENCES	1. UFSAR, Chapter 3. 2. UFSAR, Chapter 15.

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B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During movement of recently irradiated fuel assemblies within containment, a release of fission product radioactivity within containment to the environment will be restricted to the personnel air lock and containment purge and exhaust isolation valves when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths not accounted for by the Fuel Handling Accident (FHA) analysis are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to control fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." One of the containment airlocks is an integral part of the

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BASES

BACKGROUND (continued)

containment equipment hatch. During movement of recently irradiated fuel assemblies within containment, the airlock that is normally an integral part of the containment equipment hatch is typically replaced by a temporary hatch plate, which becomes an integral part of the containment equipment hatch. While the penetration plate is installed, there is only one air lock by which to enter containment. The FHA analysis assumes that the 7 ft containment personnel air lock doors are open during the accident. Closure of one of the 7 ft containment personnel air lock doors is a good practice, but not required by the FHA analysis. The analysis assumes that the equipment hatch and its associated air lock are closed. The personnel air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown the potential for containment pressurization as a result of an accident is not present, therefore, less stringent requirements are needed to isolate the containment from the environment.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment and subsequent releases to the environment is as assumed in the FHA analysis. The closure restrictions are sufficient to control fission product radioactivity release from containment due to a fuel handling accident involving handling recently irradiated fuel during refueling.

The Containment Purge and Exhaust System includes a 36 inch purge penetration and a 36 inch exhaust penetration. The purge penetration includes an 18 inch containment vacuum breaking valve, and the exhaust penetration includes an 8 inch purge bypass valve. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust flow paths are secured in the closed position. The Containment Purge and Exhaust System is not subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal 36 inch purge system is used for this purpose, and all four valves are closed manually in case of a FHA.

The 18 inch containment vacuum breaking valve is normally not used during movement of recently irradiated fuel assemblies in containment, and is maintained closed.

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BASES

BACKGROUND (continued)

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by closing a containment purge and exhaust isolation valve, or by a manual isolation valve, blind flange, or equivalent.

Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during recently irradiated fuel movements.

APPLICABLE SAFETY ANALYSES

During movement of recently irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool with both doors of the containment 7 ft personnel air lock open. The control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) and NUREG-0800, Section 6.4 (Ref. 2) are met in the case of a FHA inside containment by closing the containment purge and exhaust isolation valves in conjunction with operation of the Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System and MCR/ESGR bottled air system. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement with containment closure capability, ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 2), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the containment purge and exhaust penetrations and the 7 ft containment personnel air lock doors. A Note states that the LCO is not applicable to the 7 ft containment personnel air lock doors.

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of recently irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of recently irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident not involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time) will result in doses that are well within the guideline values specified in 10 CFR 100 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

If the containment equipment hatch or any required containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in

(continued)

BASES

ACTIONS

A.1 (continued)

the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each required containment purge and exhaust valve operator has motive power, which will ensure that each required containment purge and exhaust isolation valve is capable of being manually closed.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess of the analyses.

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation. The 18 month Frequency maintains consistency with other similar instrumentation and valve testing requirements. This Surveillance performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

BASES

REFERENCES

1. UFSAR, Section 15.4.7.
2. Standard Review Plan, Rev. 2, July 1981.
3. 10 CFR 50, Appendix A.

B 3.9 REFUELING OPERATIONS

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

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

The RHR and Coolant Circulation—High Water Level specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

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

a. Removal of decay heat;

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

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the 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.

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

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4,

(continued)

BASES

APPLICABILITY (continued)

Reactor Coolant System (RCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

BASES

ACTIONS (continued)

A.4, A.5, A.6.1, and A.6.2

If LCO 3.9.5 is not met, the following actions must be taken:

a. the equipment hatch or equipment hatch cover must be closed and secured with at least four bolts;

b. one door in each installed air lock must be closed; and

c. each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation system.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. UFSAR, Section 5.5.4.

B 3.9 REFUELING OPERATIONS

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

BASES

BACKGROUND	The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.
APPLICABLE SAFETY ANALYSES	If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge. The RHR and Coolant Circulation—Low Water Level specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).
LCO	In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide: a. Removal of decay heat;

(continued)

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 de-energized 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 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.

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)

BASES

ACTIONS

A.1 and A.2 (continued)

or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3, B.4, B.5.1, and B.5.2

If no RHR is in operation, the following actions must be taken:

a. the equipment hatch or equipment hatch cover must be closed and secured with at least four bolts;

b. one door in each installed air lock must be closed; and

c. each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation system.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described

(continued)

BASES

ACTIONS

B.3, B.4, B.5.1, and B.5.2 (continued)

above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop with the water level lowered to the level of the reactor vessel nozzles, the RHR pump net positive suction head requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.6.2

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

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

REFERENCES

1. UFSAR, Section 5.5.4.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to well below 10 CFR 100 limits.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 3).

Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.

BASES

APPLICABILITY	LCO 3.9.7 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.16, "Fuel Storage Pool Water Level."
ACTIONS	<u>A.1</u> With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.7.1</u> Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2). The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.
REFERENCES	1. Regulatory Guide 1.25, March 23, 1972. 2. UFSAR, Section 15.4.7. 3. 10 CFR 100.10.

SECTION 3.9 - REFUELING OPERATIONS

IMPROVED STANDARD TECHNICAL SPECIFICATIONS

MARKUP AND JUSTIFICATION FOR DEVIATIONS

CTS

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

3.9.1 LCO 3.9.1 Boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR. ①

APPLICABILITY: MODE 6.

------ NOTE ------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.
-------------------- TSTF-272

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.9.1 Action	A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
		AND	
		A.2 Suspend positive reactivity additions.	Immediately
		AND	
		A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
4.9.1.2	SR 3.9.1.1	Verify boron concentration is within the limit specified in COLR.	72 hours

Rev. 0

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

Primary Grade ~~Unborated~~ Water ~~Source~~ Isolation Valves 3.9.2 (1)

Flow Path

MODE 6

CTS

3.9 REFUELING OPERATIONS

3.9.2 ~~Unborated~~ Water ~~Source~~ Isolation Valves — MODE 6 (1)

3.1.1.3.2 LCO 3.9.2 Each valve used to isolate ~~unborated~~ water ~~sources~~ shall be secured in the closed position. (1)

Insert primary grade flow paths

APPLICABILITY: MODE 6.

ACTIONS

3.1.1.3.2
~~NOTE~~
~~Separate Condition entry is allowed for each unborated water source isolation valve.~~ (2)

Action

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ~~NOTE Required Action A.3 must be completed whenever Condition A is entered.~~ One or more valves not secured in closed position.	A.~~1~~ 2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.~~2~~ 3	~~Initiate actions to~~ ~~s~~Secure valve in (S) closed position.	~~Immediately~~ 15 minutes (4)
	AND		
	A.~~3~~ 4	Perform SR 3.9.1.1.	~~4~~ 1 hour~~s~~ (1) (5)

Action

A.1 Suspend positive reactivity additions. AND	Immediately

(3)

Rev. 0

INSERT

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

Primary grade water flow path isolation valves may be opened under administrative control for planned boron dilution or makeup activities.

Primary grade
Flow Path

Unborated Water Source Isolation Valves - MODE 6 ①
3.9.2

in the affected flow path

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 4.1.1.3.2	SR 3.9.2.1 Verify each valve in the affected flow path that isolates ~~unborated~~ primary grade water ~~sources~~ flow paths is locked, sealed, or otherwise secured in the closed position.	~~31 days~~ within 15 minutes following a boron dilution or makeup activity ⑥①

Rev. 0

1. The North Anna boron dilution analysis requires the primary grade water flow path isolation valves to be locked, sealed, or secured in the closed position in MODES 3, 4, 5 and 6. ITS 3.1.8, Primary Grade Water Flow Path Isolation Valves, was created to provide these requirements in MODES 3, 4, and 5. ISTS 3.9.2 is renamed to "Primary Grade Water Flow Path Isolation Valves - MODE 6" to differentiate between the titles of LCO 3.1.8 and LCO 3.9.2.

   ISTS 3.9.2 is modified to reflect the North Anna boron dilution analysis. An LCO Note is added which allows the primary grade water flow path isolation valves to be opened under administrative control for planned boron dilution or makeup activities. This is permitted under the CTS and the accident analysis.

2. The ISTS 3.9.2 "separate entry condition" note is deleted as it is not necessary and is eliminated for consistency with the CTS. Under Section 1.3, a subsequent entry into the Condition would allow the full Completion Times of 15 minutes and 1 hour from the subsequent entry to complete the Required Actions.

3. The CTS Action to suspend positive reactivity additions is added to the ISTS. This addition is appropriate as other positive reactivity additions, such as temperature changes, could mask a boron dilution event and slow operator response to terminate the event.

4. The ISTS Action to immediately initiate actions to secure the valve in a closed position is changed to be consistent with the CTS requirement to secure the valve within 15 minutes. This Completion Time is sufficient to close and lock, seal, or otherwise secure the isolation valve.

5. The ISTS requirement to verify the boron concentration is changed from a 4 hour Completion Time to the CTS 1 hour Completion Time. One hour is sufficient time to request and analyze an RCS sample to determine boron concentration.

6. The ISTS Surveillance 3.9.2.1 is changed to the CTS requirement to verify each valve in the affected flowpath that isolates primary grade water flow paths is locked, sealed, or otherwise secured in the closed position within 15 minutes following a boron dilution or makeup activity. This change is necessary as the CTS allows the isolation valves to be opened under administrative control, so more frequent verification of the valve position is necessary than the ITS Frequency of 31 days. This periodic Frequency also eliminates the need for the ISTS Condition Note which states Required Action A.3 (performance of SR 3.9.1.1) is required whenever Condition A is entered. Under the North Anna ITS, opening of an primary grade water flow path isolation valve would require performance of SR 3.9.2.1.

CTS

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.2

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE. TSTF 23 2

APPLICABILITY: MODE 6.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action	A. One [required] source range neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS. AND A.2 ~~Suspend positive reactivity additions~~	Immediately Immediately Insert	1 TSTF 286
NEW	B. Two [required] source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status. AND B.2 Perform SR 3.9.1.1.	Immediately ~~4 hours~~ ~~AND~~ Once per 12 hours ~~thereafter~~	1 TSTF 96

Rev. 0

INSERT

Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
4.9.2.c	SR 3.9.3.1	Perform CHANNEL CHECK.	12 hours	
new	SR 3.9.3.2	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months	1

Rev. 0

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

2. ISTS LCO 3.9.3 is modified by an approved TSTF- 23 and changes the LCO and Actions. This TSTF is not applicable to North Anna because the evaluation of the boron dilution accident in the safety analysis does not credit operator action to mitigate the event. Therefore, the TSTF that brackets an audible alarm or count rate is not applicable and is not incorporated.

CTS

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

3.9.4

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by {[four]} bolts; 1

TSTF-68 2 3

b. One door in each installed ~~air lock~~ closed; and

*a **

NOTES

1. Not applicable to the 7ft containment personnel air lock.

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE ~~C~~ontainment ~~P~~urge and ~~E~~xhaust ~~I~~solation ~~System~~ valve. 2

APPLICABILITY: ~~During CORE ALTERATIONS,~~ During movement of recently irradiated fuel assemblies within containment. TSTF-51

NEW

2. Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. TSTF-312

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	~~A.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	A.~~2~~ 1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

Action TSTF 51

TSTF-51

Rev. 0

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.9.4	SR 3.9.4.1 Verify each required containment penetration is in the required status.	7 days
	SR 3.9.4.2 Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal. on manual initiation	{[18]} months TSTF-284 (1) (2)

Rev. 0

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

2. The requirement to have at least one of the doors in the 7 ft personnel air lock closed is not adopted. The requirement to have the purge and exhaust valves closed automatically by an OPERABLE Containment Purge and Exhaust Isolation System is also not adopted. A Note is added stating that the containment penetration requirements are not applicable to the 7 ft containment personnel air lock. The Fuel Handling Accident (FHA) inside containment analysis assumes that both doors of the 7 ft personnel air lock are open. This feature of the analysis eliminates the need to adopt TSTF-68, "Containment Personnel Airlock Open during Fuel Movement," and the changes to SR 3.9.4.2 in TSTF-284, "Add 'Met vs. Performed' to Specification 1.4, Frequency." The control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 and NUREG-0800, Section 6.4 are met in the case of a FHA inside containment by closing the containment purge and exhaust isolation valves in conjunction with operation of the Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System and MCR/ESGR bottled air system. The existing analysis does not address the other containment penetrations, so the requirements for the remaining penetrations are retained.

3. One of the containment airlocks is an integral part of the containment equipment hatch. During movement of irradiated fuel assemblies within containment, the airlock that is normally an integral part of the containment equipment hatch is typically replaced by a temporary hatch plate, which becomes an integral part of the containment equipment hatch. While the penetration plate is installed, there is only one air lock by which to enter containment. Changes are made reflecting this design.

CTS

3.9 REFUELING OPERATIONS

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

3.9.8.1

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

not be in TSTF-153

-----NOTE-----

The required RHR loop may ~~be removed from~~ operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause ~~reduction of~~ the Reactor Coolant System boron concentration.

Action C.

(RCS), coolant with

introduction into

less than required to meet the minimum required boron concentration of LCO 3.9.1

TSTF-286

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 ~~Suspend operations involving a reduction in reactor coolant boron concentration.~~ Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	AND	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	AND	
		(continued)

Action b.

TSTF-286

Action b.

Action a

Rev. 0

CTS

ACTIONS

Action b.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	~~A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.~~	~~4 hours~~

TSTF-197

Insert

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ ~~[2800]~~ 3000 gpm.	12 hours

4.9.8.1.2

1

Rev.0

INSERT

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.4	Close equipment hatch and secure with four bolts.	4 hours
	AND		
	A.5	Close one door in each installed air lock.	4 hours
	AND		
	A.6.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR		
	A.6.2	Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

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

CTS

3.9 REFUELING OPERATIONS

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

3.9.8.2

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

Insert TSTF-349 TSTF-361

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CTS	CONDITION	REQUIRED ACTION	COMPLETION TIME
Action a	A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loops to OPERABLE status.	Immediately
		OR	
new		A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately
Action b	B. No RHR loop in operation.	B.1 ~~Suspend operations involving a reduction in reactor coolant boron concentration.~~ Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately TSTF-286
		AND	
			(continued)

Rev. 0

INSERT

-----NOTES-----

1. All RHR pumps may be de-energized for ≤15 minutes when switching from one train to another provided:

   a. The core outlet temperature is maintained > 10° F below saturation temperature;

   b. No operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration; and

   c. No draining operations to further reduce RCS volume are permitted.

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing, provided that the other loop is OPERABLE and in operation.

CTS

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
Action b.	B. (continued)	B.2 Initiate action to restore one RHR loop to operation.	Immediately	
		AND		TSTF-197
Action b		~~B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.~~	~~4 hours~~	Insert 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
4.9.8.2.2	SR 3.9.6.1 Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ ~~[2800] gpm~~ Insert 2	12 hours	1
New	SR 3.9.6.2 ---NOTE--- Not required to be performed until 24 hours after a required RHR pump is not in operation. ------ Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days	2

Rev. 0

INSERT 1

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.3	Close equipment hatch and secure with four bolts.	4 hours
	AND		
	B.4	Close one door in each installed air lock.	4 hours
	AND		
	B.5.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent.	4 hours
	OR		
	B.5.2	Verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.	4 hours

INSERT 2

a. ≥ 3000 gpm, or

b. ≥ 2000 gpm if RCS temperature ≤ 140° F and time since entry into MODE 3 ≥ 100 hours.

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

2. Consistent with TSTF-265, a Note is added to SR 3.9.6.2 which permits the performance of the SR to verify correct breaker alignment and power availability to be delayed until 24 hours after a required pump is not in operation. This provision is required because when pumps are swapped under the current requirements, the Surveillance is immediately not met on the pump taken out of operation. This change avoids entering an Action for a routine operational occurrence. The change is acceptable because adequate assurance exists that the pump is aligned to the correct breaker with power available because, prior to being removed from operation, the applicable pump had been in operation. Allowing 24 hours to perform the breaker alignment verification is acceptable because the pump was in operation, which demonstrated OPERABILITY, and because 24 hours is currently allowed by invoking SR 3.0.3. This is a new Surveillance Requirement not required in CTS 3.9.8.2.

CTS

3.9.10.1

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained ≥ 23 ft above the top of reactor vessel flange.

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~ TSTF-51
During movement of irradiated fuel assemblies within containment.

Actions

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	~~A.1 Suspend CORE ALTERATIONS.~~ ~~AND~~	~~Immediately~~ TSTF-51
	A.~~2~~ 1 Suspend movement of irradiated fuel assemblies within containment.	Immediately
	~~AND~~ ~~A.3 Initiate action to restore refueling cavity water level to within limit.~~	~~Immediately~~ TSTF-20

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CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
4.9.10.1	SR 3.9.7.1 Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

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None

SECTION 3.9 - REFUELING OPERATIONS

IMPROVED STANDARD TECHNICAL SPECIFICATIONS BASES

MARKUP AND JUSTIFICATION FOR DEVIATIONS

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 ~~of 10 CFR 50, Appendix A~~ requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration. 1

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the ~~Residual Heat Removal (RHR)~~ Low Head Safety Injection System pumps. 4

The pumping action of the ~~RHR~~ Residual Heat Removal (RHR) System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling 5

(continued)

Rev. 0

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~~ 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)

established to protect against inadvertent reactivity addition.

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. 0

BASES

LCO (continued)

$\leq$ 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, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)~~—$T_{avg} > 200°F$," and~~ ~~LCO 3.1.2, "SHUTDOWN MARGIN (SDM)—$T_{avg} \leq 200°F$,"~~ ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical. TSTF-136

Insert 1 TSTF-272

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position. Insert 2 TSTF-286

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

(continued)

Rev. 0

INSERT 1

The Applicability is modified by a Note. The Note states that the limits on boron concentration are only applicable to the refueling canal and the refueling cavity when those volumes are connected to the RCS. When the refueling canal and the refueling cavity are isolated from the RCS, no potential path for boron dilution exists.

INSERT 2

Operations that individually add limited positive reactivity (e.g., temperature fluctuations from inventory addition or temperature control fluctuations), but when combined with all other operations affecting core reactivity (e.g., intentional boration) result in overall net negative reactivity addition, are not precluded by this action.

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 in each required volume is determined periodically by chemical analysis. Insert

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.

TSTF-272

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 26.~~ UFSAR Section 3.1.22. (1)

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

Rev.0

INSERT

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.

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.1 BASES, BORON CONCENTRATION

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.

Primary Grade

Flow Path

 (1)

\- MODE 6

Flow Path

B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Isolation Valves (1)

BASES

the

BACKGROUND	During MODE 6 operations, ~~all~~ isolation valves for ~~reactor makeup~~ water sources ~~containing unborated water~~ that are connected to the Reactor Coolant System (RCS) must be closed to prevent unplanned boron dilution of the reactor coolant. The isolation valves must be secured in the closed position.
	The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition made by ~~reducing~~ the boron concentration is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution.

primary grade (1)

flow paths

locked, sealed or otherwise (1)

uncontrolled reduction of (1)

primary grade (1)

flow paths

APPLICABLE SAFETY ANALYSES	The possibility of an inadvertent boron dilution event (Ref. 1) occurring during MODE 6 refueling operations is precluded by adherence to this LCO, which requires that ~~potential dilution sources~~ be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portion of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating ~~unborated water sources~~, a safety analysis for an uncontrolled boron dilution accident ~~in accordance with the Standard Review Plan (Ref. 2)~~ is not required for MODE 6.
	The RCS boron concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~.

primary grade water flow paths

primary grade

flow paths (1)

(1)

(3)

(2)

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

LCO	This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and thus avoid a reduction in SDM.

primary grade (1)

Insert (1)

(continued)

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INSERT

For Unit 1, primary grade water flow paths may be isolated from the RCS by closing valve 1-CH-217 or 1-CH-220, , 1-CH-241, FCV-1114B and FCV-1113B. For Unit 2, primary grade water flow paths may be isolated from the RCS by closing valve 2-CH140, or 2-CH-160, 2-CH-156, FCV-2114B, and FCV-2113B.

The LCO is modified by a Note which allows the primary grade water flow path isolation valves to be opened under administrative control for planned boron dilution or makeup activities.

 (1)

BASES (continued)

APPLICABILITY

In MODE 6, this LCO is applicable to prevent an inadvertent boron dilution event by ensuring isolation of ~~all sources of unborated water~~ primary grade water flow paths to the RCS. (1)

Insert 1 ~~For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated.~~ (1)

ACTIONS

~~The ACTIONS table has been modified by a Note that allows separate Condition entry for each unborated water source isolation valve.~~ (1)

Insert 2 (1)

A.~~1~~ 2

Continuation of CORE ~~ALTERATIONS~~ primary grade is contingent upon ~~maintaining the unit in compliance with~~ flow paths this LCO. locked, sealed or otherwise With any valve used to isolate ~~unborated~~ water ~~sources~~ not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "immediately" for performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position. (1)

~~Condition A has been modified by a Note to require that Required Action A.3 be completed whenever Condition A is entered.~~ (1)

A.~~2~~ 3 (1)

Locking, sealing, or

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the ~~unborated~~ primary grade water flow path isolation valves secured closed. Securing the valves in the closed position ensures that the valves cannot be inadvertently opened. The Completion Time of ~~"immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.~~ (1) (1) (1)

15 minutes provides sufficient time to close, lock, seal, or otherwise secure the flow path isolation valve.

(continued)

Rev.0

INSERT 1

In MODES 3, 4, and 5, LCO 3.1.8, Primary Grade Water Flow Path Isolation Valves, requires the primary grade water flow paths to the RCS to be isolated to prevent an inadvertent boron dilution.

In MODES 1 and 2, the boron dilution accident was analyzed and was found to be capable of being mitigated.

INSERT 2

A.1

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the primary grade water flow path isolation valves locked, sealed, or otherwise secured closed, except as allowed under administrative control by the LCO Note. Because of the possibility of an inadvertent boron dilution, other positive reactivity additions and CORE ALTERATIONS must be prohibited while securing the isolation valves on the unborated water systems. The Completion Time of "Immediately" for suspending positive reactivity additions and CORE ALTERATIONS reflects the importance of preventing known positive reactivity additions so that any boron dilution event can be readily identified and terminated.

BASES

ACTIONS (continued)

A.~~3~~ 4 (1)

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1 (verification of boron concentration) must be performed ~~whenever Condition A is entered~~ to demonstrate that the required boron concentration exists. The Completion Time of (1) ~~4~~ hour~~s~~ is sufficient to obtain and analyze a reactor coolant sample for boron concentration. (1)

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1 (1)

These valves are to be locked, sealed, or otherwise secured closed to isolate possible dilution paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that ~~all unborated~~ the primary grade water (1) flow paths ~~sources~~ are isolated, precluding a dilution. The boron (1) concentration is checked every 72 hours during MODE 6 under SR 3.9.1.1. Insert ~~This Surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency~~ (1) This Frequency is based on engineering judgment and is considered reasonable in view of other administrative controls that will ensure that the valve opening is an unlikely possibility.

REFERENCES

1. U FSAR, Section ~~[~~15.2.4~~]~~. (3) (4)

~~2. NUREG-0800, Section 15.4.6.~~ (3)

Rev. 0

INSERT

The Frequency is based on verifying that the isolation valves are locked, sealed, or otherwise secured within 15 minutes following a boron dilution or makeup activity.

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.2 BASES, PRIMARY GRADE WATER FLOW PATH ISOLATION VALVES - MODE 6

1. The Bases to ITS 3.9.2 have been modified to reflect the changes made to the ITS.

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

TSTF 23 6

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) ~~with a [5]% instrument accuracy~~. 4 The detectors also provide continuous visual indication in the control room and an audible alarm 3 to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "~~Unborated Water Source Isolation Valves~~ Primary Grade Water Isolation Valves - MODE 6." 3

The source range neutron flux monitors satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36(c)(2)(ii). 2

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

(continued)

Rev. 0

BASES (continued)

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

TSTF 286

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and ~~positive reactivity additions~~ must be suspended immediately. <INSERT 1> Performance of Required Action A.1 shall not preclude completion <INSERT 2> of movement of a component to a safe position.

TSTF 286

B.1

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of ~~6~~ once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration ~~.~~ and ~~The Frequency of once per 12 hours~~ ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low

TSTF 96

(continued)

Rev. 0

INSERT 1

introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1

INSERT 2

Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operations.

BASES

ACTIONS

B.2 (continued)

probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant unit outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. 3

REFERENCES

1. UFSAR, Chapter 3. ~~10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29~~ 5

2. UFSAR, ~~Section~~ Chapter 15.2.4. 1 3

Rev. 0

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.3 BASES, NUCLEAR INSTRUMENTATION

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

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. 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. The specific accuracy of the Source Range channel is not a part of the licensing basis of North Anna and has been deleted.

5. 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.

6. ISTS 3.9.3 Bases are modified by an approved TSTF- 23. This TSTF is not applicable to North Anna because the evaluation of the boron dilution accident in the safety analysis does not credit operator action to mitigate the event. Therefore, the TSTF that brackets an audible alarm or count rate is not applicable and is not incorporated.

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

recently

BACKGROUND

During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted ~~from escaping~~ ~~to the environment~~ when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

TSTF-51

9

to the personnel air lock and containment purge and exhaust isolation valves

not accounted for by the Fuel Handling Accident (FHA) analysis

9

The containment serves to ~~contain~~ fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

control

9

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

recently TSTF-51

INSERT

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." ~~Each~~ air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown

6

9

The

(continued)

Rev. 0

INSERT

One of the containment airlocks is an integral part of the containment equipment hatch. During movement of recently irradiated fuel assemblies within containment, the airlock that is normally an integral part of the containment equipment hatch is typically replaced by a temporary hatch plate, which becomes an integral part of the containment equipment hatch. While the penetration plate is installed, there is only one air lock by which to enter containment. The FHA analysis assumes that the 7 ft containment personnel air lock doors are open during the accident. Closure of one of the 7 ft containment personnel air lock doors is a good practice, but not required by the FHA analysis. The analysis assumes that the equipment hatch and its associated air lock are closed.

BASES

BACKGROUND (continued)

INSERT

~~when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.~~ (9)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment ~~will be restricted from escaping~~ to the environment. The closure restrictions are sufficient to ~~restrict~~ fission product radioactivity release from containment due to a fuel handling accident during refueling. (9)

is as assumed in the FHA analysis

and subsequent release (9)

control

involving handling recently irradiated fuel (9) TSTF-51

The Containment Purge and Exhaust System ~~includes two subsystems. The normal subsystem~~ includes a ~~42~~ 36 inch purge penetration and a ~~42~~ inch exhaust penetration. ~~The second subsystem, a minipurge system, includes an 8 inch purge penetration and an 8 inch exhaust penetration.~~ During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust ~~penetrations~~ are secured in the closed position. ~~The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is~~ subject to a Specification in MODE 5. (7)

The purge penetration includes an 18 inch containment vacuum breaking valve, and the exhaust penetration includes an 8 inch purge bypass valve.

flowpaths

The Containment Purge and Exhaust System is not

In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal ~~42~~ 36 inch purge system is used for this purpose, and all four valves are closed ~~by the ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."~~ (7) (9)

Manually in case of a FHA.

~~[The minipurge system remains operational in MODE 6, and all four valves are also closed by the ESFAS.~~

~~or~~

~~The minipurge system is not used in MODE 6. All four 8 inch valves are secured in the closed position.]~~ (1)

The 18 inch containment vacuum breaking valve is normally not used during movement of recently irradiated fuel assemblies in containment, and is maintained closed.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere

(continued)

Rev. 0

INSERT

the potential for containment pressurization as a result of an accident is not present, therefore, less stringent requirements are needed to isolate the containment from the environment.

closing a containment purge and exhaust

9

BASES

BACKGROUND (continued)

must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements ~~(Ref. 1)~~.

involving handling recently irradiated fuel

3

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool ~~or a heavy object onto other irradiated fuel assemblies~~. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," ~~and the~~ in conjunction with minimum decay time of 100 hours prior to ~~CORE ALTERATIONS~~ irradiated fuel movement with containment closure capability, ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 2), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

TSTF-51

TSTF-51

1

2

INSERT 1

3 5

8 9

TSTF-51

2

3

INSERT 3

Containment penetrations satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

2

TSTF-312

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the ~~OPERABLE~~ containment purge and exhaust penetrations. ~~For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve~~

TSTF-51

9

INSERT 2

(continued)

Rev. 0

INSERT 1

with both doors of the containment 7 ft personnel air lock open. The control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 (Ref. 3) and NUREG-0800, Section 6.4 (Ref. 2) are met in the case of a FHA inside containment by closing the containment purge and exhaust isolation valves in conjunction with operation of the Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System and MCR/ESGR bottled air system.

INSERT 2

and the 7 ft containment personnel air lock doors. A Note states that the LCO is not applicable to the 7 ft containment personnel air lock.

INSERT 3

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of recently irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

BASES

LCO (continued)

~~closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.~~ 9

APPLICABILITY

recently

the limiting

The containment penetration requirements are applicable during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment because this is when there is a potential for ~~a~~ fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment ~~are~~ is not being conducted, the potential for a fuel handling accident does not exist. Insert Therefore, under these conditions no requirements are placed on containment penetration status. TSTF-51 10

ACTIONS

A.1 ~~and A.2~~ TSTF-51

required

If the containment equipment hatch, ~~air locks,~~ or any required containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, ~~including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open,~~ the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending ~~CORE ALTERATIONS and~~ movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position. 9 9 TSTF-51

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will

(continued)

Rev. 0

INSERT

Additionally, due to radioactive decay, a fuel handling accident involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within a time frame established by analysis. The term recently is defined as all irradiated fuel assemblies, until analysis is performed to determine a specific time) will result in doses that are well within the guideline values specified in 10 CFR 100 even without containment closure capability.

required containment purge and exhaust isolation

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

manually

demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed ~~by an OPERABLE automatic containment purge and exhaust isolation signal~~. (9)

involving handling recently irradiated fuel

recently

The Surveillance is performed every 7 days during ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment. TSTF-51

in excess of the analyses

significant

TSTF-51 (3)

SR 3.9.4.2

(5) TSTF-284

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation ~~or on an actual or simulated high radiation signal~~. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. ~~In LCO 3.3.6, the Containment Purge and Exhaust Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.~~ These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment. (9)

This

Rev. 0

BASES (continued)

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988. 3

1 2. U FSAR, Section [15.4.6]. 7 3 1

2 3. NUREG-0800, ~~Section~~ 15.7.4, Rev. 1, July 1981. 2 3

Standard Review Plan

3. 10 CFR 50, Appendix A. 3

Rev. 0

JUSTIFICATION FOR DEVIATIONS
ITS 3.9.4 BASES, CONTAINMENT PENETRATIONS

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

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. 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. Changes are made adding material moved to the Bases from the Technical Specifications.

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

6. One of the containment airlocks is an integral part of the containment equipment hatch. During movement of irradiated fuel assemblies within containment, the airlock that is normally an integral part of the containment equipment hatch is replaced by a temporary hatch plate, which becomes an integral part of the containment equipment hatch. While the penetration plate is installed, there is only one air lock by which to enter containment. Changes are made reflecting this design.

7. Changes are made to describe the Containment Purge and Exhaust System design because the ISTS description assumes two subsystems, which does not reflect the configuration at NAPS.

8. The reference to a dropped heavy object onto irradiated fuel assemblies in the Applicable Safety Analyses is deleted because the analysis is for a dropped irradiated fuel assembly.

9. The requirement to have at least one of the doors in the 7 ft personnel air lock closed is not adopted. The requirement to have the purge and exhaust valves closed automatically by an OPERABLE Containment Purge and Exhaust Isolation System is also not adopted. A Note is added stating that the containment penetration requirements are not applicable to the 7 ft containment personnel air lock. The Fuel Handling Accident (FHA) inside containment analysis assumes that both doors of the 7 ft personnel air lock are open. The control room operator dose limits of 10 CFR 50, Appendix A, GDC-19 and NUREG-0800, Section 6.4 are met in the case of a FHA inside containment by closing the containment purge and exhaust isolation valves in conjunction with operation of the Main Control Room/Emergency Switchgear Room (MCR/ESGR) Emergency Ventilation System and MCR/ESGR bottled air system. The existing analysis does not address the other containment penetrations, so the requirements for the remaining penetrations are

retained. The Bases are modified to describe the allowance to have the 7 ft containment personnel air lock open, and account for manual closure of the containment purge and exhaust valves.

10. TSTF-51 revises the ISTS to only apply Technical Specification controls during movement of "recently" irradiated fuel. A fuel handling accident without containment closure with fuel that has not been irradiated "recently" will not result in offsite doses that exceed the guidelines in 10 CFR 100. Therefore, the containment closure requirements are only required when moving recently irradiated fuel assemblies. The proposed Bases in TSTF-51 defines "recently" irradiated fuel as fuel that has been part of a critical reactor core within a licensee-specified number of days. The Company has not determined a plant-specific value for this decay time. Therefore, the Bases are modified to state that until analyses are performed to determine a specific value, all irradiated fuel assemblies will be considered "recently irradiated." This change is appropriate because it maintains Technical Specifications controls on all irradiated fuel and provides the ability to establish a specific decay time as the definition of "recently" irradiated under the Technical Specifications Bases Control Program.

B 3.9 REFUELING OPERATIONS

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

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

loop to not be in operation

removal from operation

loop

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

The RHR and Coolant Circulation- High Water level specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii)

(continued)

Rev.0

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

(2)

LCO

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

a. Removal of decay heat;

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

c. Indication of reactor coolant temperature.

RHR discharge (3)

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the ~~low end~~ temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. (3) TSTF 153

at least one of

The LCO is modified by a Note that allows the required operating RHR loop to ~~be removed from service~~ not be in operation for up to 1 hour per 8 hour period, provided no operations are permitted that would ~~cause a reduction of~~ dilute the RCS boron concentration by introduction of coolant into the RCS with boron concentration less than required to meet the minimum boron concentration of LCO 3.9.1. Boron concentration reduction with coolant at boron concentrations less than required to assure the RCS boron concentration is maintained is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity. TSTF 286

APPLICABILITY

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

(continued)

Rev.0

BASES

APPLICABILITY (continued)

requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS)~~, and Section 3.5, Emergency Core Cooling Systems (ECCS)~~. RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level." 3

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of unborated water sources are isolated.~~ Insert TSTF-286

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling

(continued)

Rev. 0

INSERT

Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin in maintaining subcritical operation.

BASES

ACTIONS

A.3 (continued)

water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4 , A.5, A.6.1, and A.6.2

TSTF-197

~~If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.~~

~~The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.~~

Insert 1

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. U FSAR, Section ~~[5.5.7]~~ 5.5.4.

③ ④

Rev. 0

INSERT

If LCO 3.9.5 is not met, the following actions must be taken:

a) the equipment hatch or equipment hatch cover must be closed and secured with at least four bolts;
b) one door in each installed air lock must be closed; and
c) each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation system.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

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

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

B 3.9 REFUELING OPERATIONS

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

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

LCO

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

(continued)

The RHR and Coolant Circulation – Low Water Level specification satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Rev. 0

BASES

LCO (continued)

Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

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

c. Indication of reactor coolant temperature.

TSTF-349
TSTF-361
Insert

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 ~~low end~~ RHR discharge 3 temperature. The flow path starts in one of the RCS hot legs and is returned to at least one of the RCS cold legs. 3

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) ~~and Section 3.5, Emergency Core Cooling Systems (ECCS)~~. 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." 3

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 or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

(continued)

Rev. 0

INSERT

This LCO is modified by two Notes. Note 1 permits the RHR pumps to be de-energized 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 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.

BASES

ACTIONS (continued)

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated.~~

Insert 1

TSTF-286

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3 , B.4, B.5.1, and B.5.2

TSTF-197

~~If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.~~

~~The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.~~

Insert 2

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. In addition, during operation of the RHR loop with the water level ~~in the vicinity~~ lowered to the level of the reactor vessel nozzles, the RHR pump net positive suction head requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control,

3

(continued)

Rev. 0

INSERT 1

Suspending positive reactivity additions that could result in failure to meet the minimum boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin in maintaining subcritical operation.

INSERT 2

If no RHR is in operation, the following actions must be taken:

a) the equipment hatch or equipment hatch cover must be closed and secured with at least four bolts;
b) one door in each installed air lock must be closed; and
c) each penetration providing direct access from the containment atmosphere to the outside atmosphere must be either closed by a manual or automatic isolation valve, blind flange, or equivalent, or verified to be capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation system.

With RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Performing the actions described above ensures that all containment penetrations are either closed or can be closed so that the dose limits are not exceeded.

The Completion Time of 4 hours allows fixing of most RHR problems and is reasonable, based on the low probability of the coolant boiling in that time.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

and alarm indications available to the operator for monitoring the RHR System in the control room.

SR 3.9.6.2

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

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

5

REFERENCES

1. FSAR, Section ~~[5.5.7]~~ 5.5.4

3 4

Rev. 0

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

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

5. Consistent with TSTF-265, a Note is added to SR 3.9.6.2 which permits the performance of the SR to verify correct breaker alignment and power availability to be delayed until 24 hours after a required pump is not in operation. This provision is required because when pumps are swapped under the current requirements, the Surveillance is immediately not met on the pump taken out of operation. This change avoids entering an Action or invoking SR 3.0.3 for a routine operational occurrence. The change is acceptable because adequate assurance exists that the pump is aligned to the correct breaker with power available because, prior to being removed from operation, the applicable pump had been in operation. Allowing 24 hours to perform the breaker alignment verification is acceptable because the pump was in operation, which demonstrated OPERABILITY, and because 24 hours is currently allowed by invoking SR 3.0.3. This is a new Surveillance Requirement not required in CTS 3.9.8.2.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies ~~or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~ within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~< 25% of~~ Well below 10 CFR 100 limits~~, as provided by the guidance of Reference 3~~. TSTF-51 2 1

APPLICABLE SAFETY ANALYSES

During ~~CORE ALTERATIONS and~~ movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1). TSTF-51

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft ~~and a minimum decay time of 100 hours prior to fuel handling~~, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref~~s~~. ~~4~~ 3 ~~and 5~~). 6 1 3

(continued)

Rev. 0

BASES

APPLICABLE SAFETY ANALYSES (continued)

Refueling cavity water level satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (5)

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits~~, as provided by the guidance of Reference 3~~. (1)

APPLICABILITY

LCO 3.9.7 is applicable ~~during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and~~ when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.~~15~~ 16, "Fuel Storage Pool Water Level." TSTF-51 (2)

ACTIONS

A.1 ~~and A.2~~

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving ~~CORE ALTERATIONS or~~ movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur. TSTF-51

The suspension of ~~CORE ALTERATIONS and~~ fuel movement shall not preclude completion of movement of a component to a safe position. TSTF-51

~~A.3~~

~~In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, action to restore refueling cavity water level must be initiated immediately.~~ TSTF-20

(continued)

Rev. 0

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
2. u FSAR, Section ~~[15.4.5]~~. 15.4.7 ② ④
~~3. NUREG-0800, Section 15.7.4.~~ ①
③ ~~4.~~ 10 CFR 100.10. ①
~~5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.~~ ③

Rev. 0

1. North Anna Units 1 and 2 are not committed to the Standard Review Plan (NUREG-0800). References to NUREG-0800 have been eliminated and subsequent references have been renumbered.

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. A reference to WCAP-828 is eliminated. The document is not part of the North Anna licensing basis.

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

5. 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.

6. The minimum decay time described in the Bases is deleted. The CTS decay time specification is relocated to the Technical Requirements Manual. Retaining the value in the Bases is confusing and may result in inconsistency.

SECTION 3.9 - REFUELING OPERATIONS

CURRENT TECHNICAL SPECIFICATIONS

MARKUP AND DISCUSSION OF CHANGES

UNIT 1

A.1

ITS 3.9.1

4-14-87

ITS

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

and the refueling cavity A.3

LIMITING CONDITION FOR OPERATION

3.9.1 3.9.1 ~~With the reactor vessel head unbolted or removed,~~ the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained ~~uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:~~ A.2

a. ~~Either a K_{eff} of 0.95 or less, or~~ LA.1

b. ~~A boron concentration of ≥ 2300 ppm~~ within the limit in the COLR

Applicability Note

APPLICABILITY: MODE 6~~*~~. Note — Only applicable to the refueling canal and refueling cavity when connected to the RCS. A.2 L.3

Condition A

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration ~~at ≥ 10 gpm of ≥ 12,950 ppm boric acid solution or its equivalent~~ ~~until K_{eff} is reduced to ≤ 0.95 or the boron concentration is restored to ≥ 2300 ppm, whichever is the more restrictive.~~ ~~The provisions of Specification 3.0.3 are not applicable.~~ L.1 LA.1 A.4

until boron concentration is within limit

SURVEILLANCE REQUIREMENTS

~~4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:~~ L.2

~~a. Removing or unbolting the reactor vessel head, and~~

~~b. Withdrawal of any full length control rod located within the reactor pressure vessel, in excess of 3 feet from its fully inserted position.~~

SR 3.9.1.1 4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

~~* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.~~ A.2

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UNIT 2

A.1

ITS 3.9.1

4-14-87

ITS

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

and the refueling cavity A.3

LIMITING CONDITION FOR OPERATION

3.9.1 3.9.1 ~~With the reactor vessel head unbolted or removed,~~ the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained ~~uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:~~ A.2

~~a. Either a K_{eff} of 0.95 or less, or~~ LA.1

~~b. A boron concentration of $\geq$ 2300 ppm~~ within the limit in the COLR A.2

Applicability Note APPLICABILITY: Mode 6~~*~~

NOTE
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

L.3

ACTION:

Condition A With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration ~~at $\geq$ 10 gpm of $\geq$ 12,950 ppm boric acid solution or its equivalent until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ 2300 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.~~ L.1 A4

until boron concentration is within limit. LA.1

SURVEILLANCE REQUIREMENTS

~~4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:~~ L.2

~~a. Removing or unbolting the reactor vessel head, and~~

~~b. Withdrawal of any full length control rod located within the reactor pressure vessel, in excess of 3 feet from its fully inserted position.~~

SR 3.9.1.1 4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

~~* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.~~ A.2

Page 1 of 1 Rev 0

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 LCO 3.9.1 states that with the reactor vessel head unbolted or removed, the boron concentration must be within the limit provided in the LCO. The CTS 3.9.1 Applicability is modified by a footnote that states, "The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed." ITS 3.9.1 does not include the phrase "with the reactor vessel head unbolted or removed" or the Applicability footnote.

This change is acceptable because the technical requirements have not changed. Both the ITS and CTS Specifications are applicable in MODE 6. The ITS defined MODE 6 as, "one or more reactor vessel head closure bolts less than fully tensioned." Therefore, the CTS LCO statement is equivalent to the ITS Applicability and the conditions under which the LCO applies have not changed. The ITS MODE 6 Applicability is defined as the reactor vessel head unbolted or removed, so the Applicability footnote is not required. This change is designated as administrative because the technical requirements of the specifications have not changed.

A.3 CTS 3.9.1 provides requirements on the boron concentration of filled portions of the Reactor Coolant System and the refueling canal. The ITS provides requirements on the boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity.

This change is acceptable because the technical requirements have not changed. The refueling cavity is considered to be governed by the CTS requirements because the refueling cavity is typically connected to the RCS, the refueling canal, or both. This change is designated as administrative because the technical requirements of the specifications have not changed.

A.4 CTS 3.9.1 Action contains the statement, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.1 does not contain an equivalent statement.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the LCO 3.0.3 exception is not needed. This change is designated as administrative because the technical requirements of the specifications have not changed.

MORE RESTRICTIVE CHANGES

None

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* CTS 3.9.1 states that the boron concentration in MODE 6 shall be the more restrictive of a K_{eff} of 0.95 or a boron concentration of ≥ 2300 ppm. ITS LCO 3.9.1 states that the boron concentration shall be within the limit specified in the COLR. This changes the CTS by relocating the MODE 6 boron concentration limit to the Core Operating Limits Report (COLR).

The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits From the Technical Specifications, that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. ITS 3.9.1 continues to require that boron concentration limit is met. SR 3.9.1.1 requires periodic verification that boron concentration is within the limits provided in the COLR. The method of determining or utilizing the boron concentration limit has not changed. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, Core Operating Limits Report. ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.9.1 ACTION states that when the boron concentration requirement is not met, initiate and continue boration at ≥ 10 gpm of ≥ 12,950 ppm boric acid solution or its equivalent until K_{eff} is reduced to ≤ 0.95 or the boron concentration is restored to ≥ 2300 ppm, whichever is more restrictive. ITS 3.9.1 requires initiation of action to restore boron concentration to within limit. This changes the CTS by eliminating the specific requirements for the boric acid solution to be used to restore compliance with the LCO.

The purpose of the CTS 3.9.1 Action is to preclude a reactivity event while the boron concentration is below the limit. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Specifying the boric acid solution requirements in the Action is not necessary, as the ITS requires that action to restore the boron concentration immediately. This will result in the boron concentration being restored as quickly, or more quickly, than the CTS requirement. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 5 – Deletion of Surveillance Requirement)* CTS 4.9.1.1 requires the LCO reactivity condition to be determined prior to removing or unbolting the reactor vessel head, and prior to withdrawal of any full length control rod located within the reactor pressure vessel, in excess of 3 feet from its fully inserted position. ITS 3.9.1 does not contain this Surveillance Requirement.

The purpose of CTS 3.9.1.1 is to ensure that the MODE 6 requirements are met prior to entering MODE 6 and that the reactor has sufficient SHUTDOWN MARGIN prior to withdrawing any control rods. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the values used to meet the LCO are consistent with the safety analysis Thus, appropriate values continue to be tested in a manner and at a frequency necessary to give confidence that the assumptions in the safety analysis are protected. ITS 3.9.1 requires that the boron concentration be met in MODE 6 or that action be initiated to restore the boron concentration immediately and that all positive reactivity additions be suspended. Therefore, verification that the boron concentration requirement is met must be performed prior to entering MODE 6 in order to avoid immediately entering into an Action and withdrawal of control rods is prohibited when the boron concentration requirement is not met. While the CTS Surveillance is not required, the level of protection provided is appropriate. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.3 *(Category 2 – Relaxation of Applicability)* CTS 3.9.1 provides limits on the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal. ITS 3.9.1 modifies this requirement with a Note which states, "Only applicable to the refueling canal and refueling cavity when connected to the RCS." This changes the CTS by eliminating the applicability of the boron concentration limits on the refueling canal and refueling cavity when those volumes are not connected to the RCS.

The purpose of CTS 3.9.1 is to ensure the boron concentration of the water surrounding the reactor fuel is sufficient to maintain the required shutdown margin. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. If the refueling canal and refueling cavity are not connected to the RCS (such as when the reactor vessel head is on the reactor vessel), the boron concentration of those volumes cannot affect the shutdown margin of the core. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

ITS 3.9.2, PRIMARY GRADE WATER FLOW PATH ISOLATION VALVES - MODE 6

UNIT 1

A.1

ITS

REACTIVITY CONTROL SYSTEMS

BORON DILUTION

VALVE POSITION

Insert proposed LCO 3.9.2
Insert proposed LCO 3.9.2 Note

LIMITING CONDITION FOR OPERATION

LCO 3.9.2

~~3.1.1.3.2 The following valves shall be locked, sealed or otherwise secured in the closed position except during planned boron dilution or makeup activities~~ A.2

~~a. 1-CH-217 or~~

~~b. 1-CH-220, 1-CH-241, FCV-1114B and FCV-1113B.~~ LA.2

APPLICABILITY: MODES 3, 4, 5, and 6 <See ITS 3.1.8>

ACTION:

With the above valves not locked, sealed or otherwise secured in the closed position:

a. In MODES 3 and 4 be in COLD SHUTDOWN within 30 hours <See ITS 3.1.8>

Action A.1, A.2, A.3

b. In MODES 5 and 6 suspend all operations involving positive reactivity changes or CORE ALTERATIONS and lock, seal or otherwise secure the valves in the closed position within 15 minutes.

Action A.4 Insert Proposed ITS 3.9.2, Action A.4 M.1

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

4.1.1.3.2 The above listed valves shall be verified to be locked, sealed or otherwise secured in the closed position within 15 minutes after a planned boron dilution or makeup activity.

ITS 3.9.2, PRIMARY GRADE WATER FLOW PATH ISOLATION VALVES - MODE 6

UNIT 2

A.1

8-27-90

REACTIVITY CONTROL SYSTEM

ITS

BORON DILUTION

VALVE POSITION

Insert proposed LCO 3.9.2
Insert proposed LCO 3.9.2 Note

LIMITING CONDITION FOR OPERATION

LCO 3.9.2

~~3.1.1.3.2 The following valves shall be locked, sealed or otherwise secured in the closed position except during planned boron dilution or makeup activities:~~ A.2

a. ~~2-CH-140 or~~

b. ~~2-CH-160, 2-CH-156, FCV-2114B and FCV-2113B.~~ LA.2

APPLICABILITY: MODES 3, 4, 5, and 6. <See ITS 3.1.8>

ACTION:

Action A.1, A.2, A.3, A.4

With the above valves not locked, sealed or otherwise secured in the closed position: 1) suspend all operations involving positive reactivity changes or CORE ALTERATIONS, 2) lock, seal or otherwise secure the valves in the closed position within 15 minutes, ~~and 3) verify that the SHUTDOWN MARGIN is greater than or equal to 1.77% delta k/k within 60 minutes.~~ LA.1

Perform SR 3.9.1.1 within 1 hour

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

4.1.1.3.2 The above listed valves shall be verified to be locked, sealed or otherwise secured in the closed position within 15 minutes after a planned boron dilution or makeup activity.

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 3.1.1.3.2 states, "The following valves shall be locked, sealed or otherwise secured in the closed position except during planned boron dilution or makeup activities." ITS LCO 3.9.2 states, "Each valve used to isolate primary grade water flow paths shall be secured in the closed position." A Note to the LCO states, "Primary grade water flow path isolation valves may be opened under administrative control for planned boron dilution or makeup activities." ITS SR 3.9.2.1 states, "Verify each valve that isolates primary grade water flow paths is locked, sealed, or otherwise secured in the closed position."

This change is acceptable because the technical requirements have not changed. In the ITS, requirements that valves be locked, sealed, or otherwise secured are located in the Surveillances, not the LCO. Under SR 3.0.1, the SRs provide requirements necessary to meet the LCO. Therefore, moving the requirement from the LCO to the SR has no effect. The addition of the phrase "under administrative control" to the LCO Note is consistent with the ITS conventions and does not change the application of the Note as, according to UFSAR Section 15.2.4, strict administrative controls are applied to the operation of the primary grade water flow path isolation valves. This change is designated as administrative because it does not result in a technical change to the specifications.

MORE RESTRICTIVE CHANGES

M.1 Unit 1 CTS 3.1.1.3.2 states that when the primary grade water flow path isolation valves are not locked, sealed, or otherwise secured in the closed position in MODE 6, all operations involving positive reactivity changes or CORE ALTERATIONS must be suspended, and the valves must be locked, sealed, or secured in the closed position within 15 minutes. Unit 2 CTS 3.1.1.3.2 states that when the primary grade water flow path isolation valves are not locked, sealed, or otherwise secured in the closed position, all operations involving positive reactivity changes or CORE ALTERATIONS must be suspended, the isolation valves must be locked, sealed, or otherwise secured in the closed position within 15 minutes, and SHUTDOWN MARGIN must be verified greater than or equal to 1.77% $\Delta k/k$ within 60 minutes. ITS 3.9.2 Actions state than when one or more valves are not secured in the closed position, positive reactivity additions and CORE ALTERATIONS must be suspended immediately, the primary grade water flow paths must be isolated within 15 minutes and the boron concentration must be verified per SR

3.9.1.1 within 1 hour. This changes the CTS by adding a requirement to verify the RCS boron concentration within 1 hour and by changing the shutdown margin requirement from 1.77%Δk/k to a K_{eff} of 0.95.

This change is acceptable because it establishes reasonable compensatory measures for a failure to close the primary grade water flow path isolation valves. SR 3.9.1.1 requires verification that the RCS boron concentration is within the limits provided in the COLR. It is performed to verify that any inadvertent boron dilution that may have occurred has been detected and corrected. The Completion Time of 1 hour is reasonable, based on the time required to request and have analyzed an RCS water sample to determine the boron concentration. This change also makes the Unit 1 and Unit 2 requirements the same. This change is designated as more restrictive because it adds requirements to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 5 – Removal of Cycle-Specific Parameter Limits from the Technical Specifications to the Core Operating Limits Report)* Unit 2 CTS 3.1.1.3.2 Action states that with the primary grade water flow path isolation valves not locked, sealed, or otherwise secured in the closed position, verify the SHUTDOWN MARGIN is greater than or equal to 1.77% Δk/k within 60 minutes. ITS 3.9.2, Action A.4, states this requirement as, "Perform SR 3.9.1.1" within 1 hour. ITS SR 3.9.1.1 requires verification that the RCS boron concentration is within the limit provided in the COLR. This changes the CTS by moving the SHUTDOWN MARGIN value to the COLR.

The removal of these cycle-specific parameter limits from the Technical Specifications and their relocation into the COLR is acceptable because these limits are developed or utilized under NRC-approved methodologies. The NRC documented in Generic Letter 88-16, Removal of Cycle-Specific Parameter Limits From the Technical Specifications, that this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements and Surveillances that verify that the cycle-specific parameter limits are being met. ITS 3.9.1, Boron Concentration, is based on verifying that the required SHUTDOWN MARGIN is maintained in MODE 6. Also, this change is acceptable because the removed information will be adequately controlled in the COLR under the requirements provided in ITS 5.6.5, Core Operating Limits Report. ITS 5.6.5 ensures that the applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, and nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met. This change is designated as a less restrictive removal of detail change because information relating to cycle-specific parameter limits is being removed from the Technical Specifications.

LA.2 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* Unit 1 CTS 3.1.1.3.2 states "The following valves shall be locked, sealed, or otherwise secured in the closed position except during planned boron dilution or makeup activities: a. 1-CH-217 or b. 1-CH-220, 1 CH-241, FCV 1114B and FCV-1113B." Unit 2 CTS 3.1.1.3.2 states "The following valves shall be locked, sealed, or otherwise secured in the closed position except during planned boron dilution or makeup activities: a. 2-CH-140 or b. 2-CH-160, 2 CH-156, FCV 2114B and FCV-2113B." ITS 3.9.2 states, "Primary grade water flow paths shall be isolated from the RCS." ITS 3.9.2 LCO Note states, "Primary grade water flow path isolation valves may be opened under administrative control for planned boron dilution or makeup activities." This changes the CTS by relocating the list of primary grade water flow path isolation valves to the ITS Bases. The other changes in CTS 3.1.1.3.2 are discussed in DOC A.2.

The removal of these details, which are related to system design, 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 that the primary grade water flow path isolation valves be closed and the valves be verified to be locked, sealed, or otherwise secured. Listing the valves in the LCO is inconsistent with the ITS conventions. Also, this change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

None

UNIT 1

ITS

3.9

3.9.3

A.1

REFUELING OPERATIONS

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

M.4

LCO 3.9.3

OPERABLE.

3.9.2 As a minimum, two source range neutron flux monitors shall be ~~operating, each with continuous visual indication in the control room and one with audible indication in the containment.~~ LA.1

APPLICABILITY: MODE 6.

ACTION:

Action A

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or ~~positive reactivity changes~~. ~~The provisions of Specification 3.0.3 are not applicable.~~ L.1 A.2 L.1

Insert proposed Required Action A.2

Action B

M.1

INSERT proposed Action B

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. ~~A CHANNEL FUNCTIONAL TEST at least once per 7 days, and~~ L.2

b. ~~A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and.~~

SR 3.9.3.1

c. A CHANNEL CHECK at least once per 12 hours ~~during CORE ALTERATIONS.~~ M.3

SR 3.9.3.2

INSERT proposed SR 3.9.3.2 M.2

UNIT 2

A.1

ITS 3.9 REFUELING OPERATIONS

3.9.3 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

M.4

OPERABLE.

LCO 3.9.3 3.9.2 As a minimum, two source range neutron flux monitors shall be ~~operating~~, ~~each with continuous visual indication in the control room and one with audible indication in the containment.~~ LA.1

APPLICABILITY: MODE 6.

ACTION:

Action A a. With one of the above required monitors inoperable, immediately suspend all operations involving CORE ALTERATIONS ~~or positive reactivity changes~~. INSERT PROPOSED REQUIRED ACTION A.2 L.1

Action B b. With both of the above required monitors inoperable, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. ~~The provisions of Specification 3.0.3 are not applicable.~~ A.2

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1 4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours.

b. ~~A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and~~ L.2

c. ~~A CHANNEL FUNCTIONAL TEST at least once per 7 days.~~

SR 3.9.3.2 INSERT PROPOSED SR 3.9.3.2 M.2

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 3.9.2 LCO is applicable in MODE 6, but in the Action states, "The provisions of Specification 3.0.3 are not applicable." CTS LCO 3.0.3 states that the requirement is, "applicable in MODES 1, 2, 3, and 4." Therefore, LCO 3.0.3 is not applicable in MODE 6. ITS 3.9.3 does not contain this requirement. This changes the CTS by deleting the reference to LCO 3.0.3.

This change is acceptable because the statement is not required to be stated in the CTS or ITS requirements. Therefore, deleting the statement does not modify any technical requirements contained in the CTS. This change is designated as administrative because it does not result in a technical change to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS LCO 3.9.2 Action requires with less than two source range channels OPERABLE, immediate suspension of all operations involving CORE ALTERATIONS or positive reactivity changes. Unit 2 CTS in LCO 3.9.2 requires that if both monitors are inoperable, the RCS boron concentration be verified every 12 hours. ITS 3.9.3 Action A requires with one source range neutron flux monitor inoperable, CORE ALTERATIONS and reactivity changes shall be suspended immediately, "that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1." ITS Action B states with two source range neutron flux monitors inoperable, initiate action immediately to restore one to OPERABLE and perform a verification of refueling boron concentration once per 12 hours. This changes the Unit 1 CTS requirements by requiring a verification of boron concentration every 12 hours when both source ranges are inoperable and the Unit 1 and Unit 2 CTS by requiring immediate initiation action to restore one source range to OPERABLE status.

The purpose of this change is to provide necessary Required Actions that are appropriate for a possible condition that could be encountered. This change is acceptable because the proposed Required Actions are reasonable and necessary to ensure the reactor is maintained in a safe condition. This change is more restrictive because it provides for additional actions that the CTS does not require.

M.2 CTS Surveillance Requirement 4.9.2 specifies testing for the source range instrumentation channels. ITS SR 3.9.3.2 requires the performance of a CHANNEL CALIBRATION to be performed on the source range monitors every 18 months. This changes the CTS by requiring a CHANNEL CALIBRATION every 18 months on each source range monitor.

The purpose of this change is to ensure the proper testing is conducted at an appropriate frequency. This change is acceptable because a CHANNEL CALIBRATION every 18 months will continue to ensure OPERABILITY and proper operation of the source range monitors. This change is more restrictive because it provides for additional testing that the CTS does not require.

M.3 Unit 1 CTS 4.9.2 requires a CHANNEL CHECK to be performed once per 12 hours during CORE ALTERATIONS. ITS SR 3.9.3.1 requires a CHANNEL CHECK to be performed every 12 hours. This changes the Unit 1 CTS by requiring the CHANNEL CHECK to be performed every 12 hours even if CORE ALTERATIONS are not in progress.

The purpose of this change is to routinely verify the OPERABILITY of the source range monitors in conditions other than CORE ALTERATIONS. This change is acceptable because the test verifies OPERABILITY of both monitors to ensure the reactor is maintained in a safe condition. This change is more restrictive because it provides for additional testing that the CTS does not require.

M.4 CTS 3.9.2 states, in part, that, " two source range neutron flux monitors shall be operating." ITS 3.9.3 states, "Two source range neutron flux monitors shall be OPERABLE." This changes the CTS by requiring the source range neutron flux monitors to be OPERABLE, instead of just operating.

The purpose of CTS 3.9.2 is to ensure that the source range neutron flux monitors are capable of performing the safety functions assumed in the accident analysis. However, as written, the CTS LCO could be interpreted as allowing the source range neutron flux monitors to be operating in a location or condition that would prevent them from performing the assumed safety function. The ITS eliminates this possible misinterpretation. This change is acceptable because the source range neutron flux monitors must be OPERABLE, i.e., capable of performing their safety function, instead of just operating. This change is designated as more restrictive because the ITS contains more specific requirements on a component.

REMOVED DETAIL CHANGES

LA.1 *(Type 1 – Removing Details of System Design and System Description, Including Design Limits)* CTS LCO 3.9.2 states that two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room. ITS

3.9.3 LCO states that two source range neutron flux monitors shall be OPERABLE. This changes the CTS by moving the requirement that each channel has a continuous visual indication the control room and with one audible indication in the containment from the specification to the ITS Bases.

The removal of these details, which are related to system design, 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 retains the requirement that two channels to be OPERABLE and continues to require the associated testing to verify OPERABILITY. This change is acceptable because the removed information will be adequately controlled in the ITS Bases. Changes to the Bases are controlled by the Technical Specification Bases Control Program in Chapter 5. This program provides for the evaluation of changes to ensure the Bases are properly controlled. This change is designated as a less restrictive removal of detail change because information relating to system design is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.9.2 Action states that with less than two source range instrumentation channels OPERABLE, immediately suspend all operations involving positive reactivity changes. ITS 3.9.3 Action A.2 adds an allowance to this requirement, which states, "Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1." This allows positive reactivity changes provided they do not reduce the boron concentration below the refueling limit. This changes the CTS requirements by allowing a limited positive reactivity additions.

This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are 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 repair period. The requirement to maintain refueling boron concentration within limits will continue to ensure the unit will be operated within the assumptions of the safety analyses. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 5 – Deletion of Surveillance Requirement)* CTS surveillance requirement 4.9.2 states that a CHANNEL FUNCTION TEST is required for the source range neutron flux monitors at least once per 7 days and within 8 hours prior to the initial start of CORE ALTERATIONS. ITS SRs do not require the performance of similar tests for the source range instruments. This changes the CTS by deleting the CHANNEL FUNCTIONAL TESTS every 7 days and within 8 hours of CORE ALTERATIONS.

This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment used to meet the LCO is consistent with the safety analysis. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the assumptions in the safety analysis are protected. The source range instruments continue to be tested in a manner and at a frequency necessary to give confidence that the assumptions in the safety analysis are protected. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.3 *(Category 1 – Relaxation of LCO Requirements)* CTS LCO 3.9.2 states that two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment. ITS LCO 3.9.3 states that two source range neutron flux monitors shall be OPERABLE. The movement of continuous visual indication in the control room is addressed by DOC LA.1. This changes the CTS by deleting the requirement for an audible indication in the containment from the source range neutron flux monitors.

This change is acceptable because the LCO requirements continue to ensure that the source range channels are maintained consistent with the safety analyses and licensing basis. The requirement for an audible indication in the containment is not assumed by the safety analyses for core protection. The audible indication is provided to address personnel safety issues. Therefore, the audible indication is not required to be included in the Technical Specifications. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

UNIT 1

ITS

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts. A.3

b. A minimum of one door ~~in each~~ installed airlock is closed, * and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be ~~either:~~ L.1

1. Closed by an isolation valve, blind flange, or manual valve, or equivalent or LA.1

2. Be capable of being closed by an OPERABLE ~~automatic~~ Containment Purge and Exhaust isolation valve.

APPLICABILITY: During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel within the containment. L.5

ACTION:

Action A.1 With the requirements of the above specification not satisfied, immediately suspend all operations involving ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel in the containment building. ~~The provisions of Specification 3.0.3 are not applicable.~~ L.5 A.2

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either ~~in its closed/isolated condition~~ in the required status or capable of being closed by an OPERABLE ~~automatic~~ Containment Purge and Exhaust isolation valve ~~within 100 hours prior to the start of and~~ at least ~~once per 7 days~~ every 18 months during ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel in the containment building by: A.4 LA.1 L.2 L.5 L.3

SR 3.9.4.1 a. Verifying the penetrations are ~~in their closed/isolated condition~~, ~~or~~ LA.1 A.4

~~b. Testing the Containment Purge and Exhaust isolation valves and system per the applicable portions of Specifications 4.6.3.1.2 and 4.9.9.~~ LA.1

3.9.4 NOTE 1 * Both doors of the containment personnel airlock may be open provided:

~~a. One personnel airlock door is OPERABLE (i.e., the door is capable of being closed and that an individual is designated to close the door), and~~ LA.1

~~b) There is at least 23 feet of water above the top of the reactor pressure vessel flange during movement of fuel assemblies within the containment, or~~

~~b2. There is at least 23 feet of water above the top of irradiated fuel assemblies within the reactor pressure vessel during CORE ALTERATIONS excluding movement of fuel assemblies.~~ A.5

~~** If both doors of the containment personnel airlock are open pursuant to Specification 3.9.4.b above, one door shall be verified to be capable of being closed at the above surveillance frequency.~~ LA.1

3.9.4 NOTE 2 Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. L.4

A.1

ITS

SURVEILLANCE REQUIREMENTS (Continued)

b. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of the applicable cycling test and verification of isolation time.

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

See ITS 3.6.3

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

SR 3.9.4.2

c. Verifying that on a ~~Containment~~ Purge and ~~Exhaust~~ isolation signal or manual initiation, each Purge and Exhaust valve actuates to its isolation position.

LA.1

M.1

d. Cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is less than 1.2 psid and opens when the differential pressure in the direction of flow is greater than or equal to 1.2 psid but less than 5.0 psid.

See ITS 3.6.3

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

UNIT 2

ITS 3.9.4

A.1

ITS

02-27-96

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door clos~~ed and~~ held in place by a minimum of four bolts. A.3

1. b. A minimum of one door in each installed airlock is closed, * and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be ~~either:~~ L.1

1. Closed by an isolation valve, blind flange, or manual valve, or equivalent or

2. Be capable of being closed by an OPERABLE ~~automatic~~ Containment Purge and Exhaust isolation valve. LA.1

APPLICABILITY: During ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel within the containment. L.5

ACTION:

Action A.1 Action A.2 With the requirements of the above specification not satisfied, immediately suspend all operations involving ~~CORE ALTERATIONS or~~ movement of recently irradiated fuel in the containment building. ~~The provisions of Specification 3.0.3 are not applicable.~~ L.5 A.2

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either ~~in its closed/isolated condition~~ in the required status or capable of being closed by an OPERABLE ~~automatic~~ Containment Purge and Exhaust isolation valve ~~within 100 hours prior to the start of and~~ at least ~~once per 7 days~~ every 18 months ~~during CORE ALTERATIONS or~~ movement of recently irradiated fuel in the containment building by: A.4 LA.1 L.2 L.5 L.3 LA.1

SR 3.9.4.1 a. Verifying the penetrations are ~~in their closed/isolated condition~~ in the required status, ~~**~~ or A.4

3.9.4 NOTE 1 * Both doors of the containment personnel airlock may be open ~~provided:~~ LA.1

~~a. One personnel airlock door is OPERABLE (i.e., the door is capable of being closed and that an individual is designated to close the door), and~~

~~b1. There is at least 23 feet of water above the top of the reactor pressure vessel flange during movement of fuel assemblies within the containment, or~~

~~b2. There is at least 23 feet of water above the top of irradiated fuel assemblies within the reactor pressure vessel during CORE ALTERATIONS excluding movement of fuel assemblies.~~ A.5

~~** If both doors of the containment personnel airlock are open pursuant to Specification 3.9.4.b above, one door shall be verified to be capable of being closed at the above surveillance frequency.~~ LA.1

3.9.4 NOTE 2 Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. L.4

ITS 3.9.4

A.1

ITS

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

SURVEILLANCE REQUIREMENTS (Continued)

b. Testing the Containment Purge and Exhaust isolation valves and system per the applicable portions of Specifications 4.6.3.1.2 and 4.9.9. LA.1

ITS

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

See ITS 3.6.3

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

SR 3.9.4.2 c. Verifying that ~~on a Containment Purge and Exhaust isolation signal~~, each Purge and Exhaust valve actuates to its isolation position. LA.1

d. Cycling each weight or spring loaded check valve not testable during plant operation, through one complete cycle of full travel and verifying that each check valve remains closed when the differential pressure in the direction of flow is less than 1.2 psid and opens when the differential pressure in the direction of flow is greater than or equal to 1.2 psid but less than 5.0 psid.

See ITS 3.6.3

4.6.3.1.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

on manual initiation

M.1

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 3.9.4 and CTS 3.9.9 Action states, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.4 does not include this statement. ITS LCO 3.0.3 states, "LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4." This changes CTS by deleting an allowance already provided in a different portion of the ITS.

This change is acceptable because ITS LCO 3.0.3 requirements are consistent with those states in the CTS. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 3.9.4.b states, "A minimum of one door in each airlock is closed,* and." ITS 3.9.4.b states, "A minimum of one door in each installed air lock is closed, and." This changes the CTS by adding the word "installed," clarifying that the requirement applies to each airlock actually acting as part of the containment boundary, because one of the air locks is normally removed during refueling outages. The "*" is addressed by DOC L.4.

This change is acceptable because it clarifies that only the air locks actually acting as a containment boundary need to meet the LCO requirements, consistent with how the CTS requirement is implemented. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 4.9.4 requires each required containment penetration, except for those capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve, be determined in its closed/isolated position. ITS SR 3.9.4.1 states, "Verify each required containment penetration is in the required status," which is closed for each of the required containment penetrations. This changes the CTS by moving the reference to the required position of the containment penetrations from the Surveillance Requirement to the LCO. Changes associated with containment purge and exhaust isolation valves are addressed by DOC LA.1.

This change is acceptable because the containment penetrations are still required to be closed for the required penetrations, the reference to the position is moved from one part of the specification to another. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.9.4 states, "The containment building penetrations shall be in the following status:...c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either: 1. Closed by an isolation valve, blind flange, or manual valve, or 2. Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust valve." CTS 4.6.3.1.2.c requires, "Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position." CTS 4.9.4.b requires, "Testing the Containment Purge and Exhaust isolation valves and system per the applicable portions of Specifications 4.6.3.1.2 and 4.9.9." CTS 4.9.4 requires each required containment penetration, except for those capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve, be determined in its closed/isolated position. ITS 3.9.4 states, "The containment penetrations shall be in the following status:... c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either: 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or 2. Shall be containment purge and exhaust valves capable of being closed." This changes the CTS by requiring the containment purge and exhaust valves be capable of being manually closed, instead of automatically being closed. The requirement for automatic actuation is addressed by DOC LA.1.

The removal of these details, which are related to system operation, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements for all the required containment penetrations to be closed in case of a FHA. This change moves the requirements associated with maintaining the purge and exhaust isolation valves or the 7 ft personnel air lock doors closed or capable of being closed to the Technical Requirements Manual (TRM). The FHA analysis assumes that both doors of the 7 ft personnel air lock are open, and that the entire radioactive material release from the refueling cavity water to the containment air space is discharged through the purge and exhaust valves via ventilation stacks with no credit for isolation or iodine filtration. The analysis did not address having the other containment penetrations open in case of a FHA, and the requirements for these penetrations are retained. Also, this change is acceptable because the removed information will be adequately controlled in the TRM. Any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system operation is being removed from the Technical Specifications.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

LA.1 *(Type 2 – Removing Descriptions of System Operation)* CTS 3.9.4 states, "The containment building penetrations shall be in the following status:...b. A minimum of one door in each airlock is closed,* and c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either: 1. Closed by an isolation valve, blind flange, or manual valve, or 2. Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust valve." CTS 4.6.3.1.2.c requires, "Verifying that on a Containment Purge and Exhaust isolation signal, each Purge and Exhaust valve actuates to its isolation position." CTS 4.9.4.b requires, "Testing the Containment Purge and Exhaust isolation valves and system per the applicable portions of Specifications 4.6.3.1.2 and 4.9.9." CTS 3.9.4 footnote "*" states, "Both doors of the containment personnel airlock may be open provided: a. One personnel airlock door is OPERABLE (i.e., the door is capable of being closed and that an individual is designated to close the door), and." CTS 4.9.4 requires each required containment penetration, except for those capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve, be determined in its closed/isolated position. CTS 4.9.4.a footnote "**" states, "If both doors of the containment personnel airlock are open pursuant to Specification 3.9.4.b above, one door shall be verified to be capable of being closed at the above surveillance frequency." ITS 3.9.4 states, "The containment penetrations shall be in the following status:...b. One door in each installed air lock closed; and c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere closed by a manual or automatic isolation valve, blind flange, or equivalent." ITS 3.9.4 NOTE 1 states, "Not applicable to the 7 ft containment personnel air lock or the containment purge and exhaust isolation valves." This changes the CTS by moving the requirements to close or be able to close the containment personnel air lock doors and the containment purge and exhaust valves to the TRM. The change adding "installed" to the phrase in ITS 3.9.4.b is addressed by DOC A.3.

The removal of these details, which are related to system operation, from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The ITS still retains requirements for all the required containment penetrations to be closed in case of a FHA. This change moves the requirements associated with maintaining the purge and exhaust isolation valves or the 7 ft personnel air lock doors closed or capable of being closed to the Technical Requirements Manual (TRM). The FHA analysis assumes that both doors of the 7 ft personnel air lock are open, and that the entire radioactive material release from the refueling cavity water to the containment air space is discharged through the purge and exhaust valves via ventilation stacks with no credit for isolation or iodine filtration. The analysis did not address having the other containment penetrations open in case of a FHA, and the requirements for these penetrations are retained. Also, this change is acceptable because the removed information will be adequately controlled in the TRM. The TRM is incorporated by reference into the UFSAR and any changes to the TRM are made under 10 CFR 50.59, which ensures changes are properly evaluated. This change is designated as a less restrictive removal of detail change because information relating to system operation is being removed from the Technical Specifications.

LESS RESTRICTIVE CHANGES

L.1 *(Category 1 – Relaxation of LCO Requirements)* CTS 3.9.4.c.1 states that one option for the status of a containment penetration is, "Closed by an isolation valve, blind flange, or manual valve." ITS 3.9.4.c. states that one option for the status of a containment penetration is, "Closed by a manual or automatic isolation valve, blind flange, or equivalent." This changes the CTS by adding the option of having, "or equivalent," as the means of closing the penetration.

The purpose of CTS and ITS 3.9.4 is to provide assurance of containment closure. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The option of using an equivalent means of containment penetration isolation is added, which is described in the Bases. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.9.4 and 4.9.9 state that specified containment penetration surveillances shall be performed, "within 100 hours prior to the start of and at least once per 7 days during…" the specified conditions. ITS SR 3.9.4.1 do not include the, "within 100 hours prior to the start of" frequency. ITS SR 3.0.1 states, "SRs shall be met during the MODES or other specified conditions in the Applicability for the individual LCOs, unless otherwise stated in the SR." This changes the CTS by only requiring the surveillances be met within their specified frequency, not within 100 hours prior to entering the MODE of applicability.

The purpose of CTS 4.9.4 is to verify the equipment required to meet the LCO is OPERABLE. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. For CTS 4.9.4, the Surveillance Frequency of 7 days verifying containment penetrations are in the required status is acceptable during the MODE of applicability, and is also acceptable during the period prior to entering the MODE of applicability.

This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.3 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.9.4 include surveillance frequencies of once per 7 days during specified times in the MODE of applicability for testing Containment Purge and Exhaust System OPERABILITY. ITS SR 3.9.4.2 for the same requirement is 18 months. This changes the CTS by changing the Surveillance Frequency from 7 days to 18 months.

The purpose of CTS 4.9.4 is to verify the equipment required to meet the LCO is OPERABLE. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Containment Purge and Exhaust System testing is still required, but at a frequency consistent with the frequency used for containment isolations valves required in MODES 1, 2, 3, and 4, that still provides an appropriate degree of assurance that the system is OPERABLE. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.4 *(Category 1 – Relaxation of LCO Requirements)* ITS 3.9.4 Note 2 states, "Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls." CTS 3.9.4 does not include such an allowance. This changes the CTS by allowing containment penetration flow paths to be unisolated under administrative controls during movement of recently irradiated fuel assemblies.

The purpose of CTS 3.9.4 is to ensure all required containment penetrations are closed in case of a Fuel Handling Accident (FHA). This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The LCO Note requires that unisolated containment penetration flow paths be under administrative controls to ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during movement of recently irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a FHA. This provides assurance that all required penetrations are closed in case of a FHA. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

L.5 *(Category 2 – Relaxation of Applicability)* CTS 3.9.4 is applicable during CORE ALTERATIONS and movement of irradiated fuel assemblies. ITS 3.9.4 is applicable during movement of recently irradiated fuel assemblies. References to CORE ALTERATIONS in CTS 3.9.4 are eliminated in the Applicability, Action, and Surveillances. All references in CTS 3.9.4 to irradiated fuel are changed to "recently" irradiated fuel. This changes the CTS by eliminating requirements for containment closure during CORE ALTERATIONS and movement of fuel that is not recently irradiated.

The purpose of CTS 3.9.4 is to ensure that the initial assumptions of a fuel handling accident are met. Specifically, containment closure is required during CORE ALTERATIONS and movement of irradiated fuel to ensure that the offsite doses resulting from a fuel handling accident are within regulatory guidelines. This change is acceptable because the requirements continue to ensure that the structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis The only accident postulated to occur during CORE ALTERATIONS which results in significant radioactive release is a fuel handling accident. Therefore, imposing requirements during CORE ALTERATIONS and during movement of irradiated fuel assemblies is repetitive and unnecessary. Fuel handling accidents involving irradiated fuel that has not been recently irradiated will not result in offsite doses in excess of the guidelines in 10 CFR Part 100, even without containment closure. Recently irradiated fuel is defined by the decay time since the fuel has been part of a critical reactor core. The Company has not determined this plant-specific value for North Anna. Therefore, the Bases state that "recently irradiated" fuel is all irradiated fuel, until such time as the appropriate analyses are performed and the Bases modified in accordance with the Technical Specifications Bases Control Program. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

UNIT 1

A.1

ITS 3.9.5

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION

NORMAL WATER LEVEL

LIMITING CONDITIONS FOR OPERATION

ITS

3.9.5

3.9.8.1 At least one RHR loop shall be OPERABLE* and at least one RHR loop shall be in operation. A.2

APPLICABILITY: MODE 6 With the reactor vessel water level greater than or equal to 23 feet above the top of the reactor pressure vessel flange.

Action A.3

A.2

A.1

A.4-A.6

LCO NOTE

ACTION: a. With less than one RHR loop OPERABLE, immediately initiate corrective actions to return the required RHR loops to OPERABLE status as soon as possible. L.4

Insert proposed Action A.1

b. With less than one RHR loop in operation, except as provided in c. below, ~~suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.~~

Insert proposed Action A.2 A.3

Insert proposed Actions A.4, A.5, A.6.1, and A.6.2 L.1

Insert proposed LCO Note

c. ~~The RHR loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.~~ M.1

d. ~~The provisions of Specification 3.0.3 are not applicable.~~ A.4

SURVEILLANCE REQUIREMENTS

~~4.9.8.1.1 Verify the required RHR loop to be OPERABLE per Specification 4.0.5.~~ L.3

SR 3.9.5.1

4.9.8.1.2 At least once per ~~4~~ 12 hours, verify at least one RHR Loop is in operation and, L.2

~~a. if the RCS temperature >140° F or the time since entry into MODE 3 is <100 hours,~~ circulating reactor coolant at a flow rate ≥3000 gpm.

~~b. if the RCS temperature ≤140° F and the time since entry into MODE 3 is ≥100 hours, circulating reactor coolant at a flow rate ≥2000 gpm to remove decay.~~ M.2

~~* The normal or emergency power source may be inoperable for each RHR loop.~~ A.2

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UNIT 2

A.1

ITS 3.9.5

8-27-90

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION

NORMAL WATER LEVEL

ITS

LIMITING CONDITION FOR OPERATION

3.9.5

3.9.8.1 At least one RHR loop shall be OPERABLE* and at least one RHR loop shall be in operation. A.2

APPLICABILITY: MODE 6 With the reactor vessel water level greater than or equal to 23 feet above the top of the reactor pressure vessel flange. L.4

Insert proposed Action A.2 A3

Action A.3

ACTION: a. With less than one RHR loop OPERABLE, immediately initiate corrective actions to return the required RHR loops to OPERABLE status as soon as possible.

A.2

A.1

A.4-A.6

LCO Note

Insert proposed Action A.1

b. With less than one RHR loop in operation, except as provided in c. below, ~~suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.~~ L.1

Insert proposed Actions A.4, A.5, A.6.1 and A.6.2

Insert proposed LCO Note

~~c. The RHR loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.~~ M.1

~~d. The provisions of Specification 3.0.3 are not applicable.~~ A.4

SURVEILLANCE REQUIREMENTS

~~4.9.8.1.1 Verify the required RHR loop to be OPERABLE per Specification 4.0.5.~~ L.3

SR 3.9.5.1

4.9.8.1.2 At least once per ~~4~~ 12 hours, verify at least one RHR Loop is in operation and, L.2

~~a. if the RCS temperature >140° F or the time since entry into MODE 3 is <100 hours,~~ circulating reactor coolant at a flow rate ≥3000 gpm.

~~b. if the RCS temperature ≤140° F and the time since entry into MODE 3 is ≥100 hours, circulating reactor coolant at a flow rate ≥2000 gpm to remove decay heat.~~ M.2

~~* The normal or emergency power source may be inoperable for each RHR loop.~~ A.2

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 3.9.8.1 LCO is modified by a footnote, *, which states that the normal or emergency power source may be inoperable for each RHR loop. ITS 3.9.5 does not include this statement. The ITS definition of "OPERABLE" states that a component is OPERABLE if either the normal or emergency power source is OPERABLE. This changes CTS by deleting an allowance already provided in a different portion of the ITS.

This change is acceptable because the ITS definition of OPERABLE contains the necessary requirements for a component to perform its safety function. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 3.9.8.1, Action b, states, in part, that with less than one RHR loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. ITS 3.9.5 states that with the RHR loop requirements not met, suspend operations involving a reduction in reactor coolant boron concentration and suspend loading irradiated fuel assemblies in the core. This changes the CTS by requiring that the loading of irradiated fuel assembles be suspended instead of requiring that all operations involving an increase in the reactor decay heat load be suspended.

This change is acceptable because the requirements have not changed. The reactor decay heat load is generated by irradiated fuel. The only method of increasing the decay heat load of a reactor in MODE 6 is to load additional irradiated fuel assemblies into the core. Therefore, the CTS and ITS requirements are equivalent. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 3.9.8.1 Action d. states, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.5 does not include this statement. ITS LCO 3.0.3 states, "LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4." This changes CTS by deleting an allowance already provided in a different portion of the ITS.

This change is acceptable because ITS LCO 3.0.3 requirements are consistent with those stated in the CTS. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.9.8.1, Action c., states that the RHR loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs. ITS LCO 3.9.5 Notes states that the required RHR loop may not be in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction into the Reactor Coolant System, coolant with boron concentration less than required to meet the minimum required boron concentration of LCO 3.9.1. This results in two changes to the CTS. First, the allowance to remove RHR from operation is no longer restricted to CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs. Second, the use of the allowance in the ITS is predicated on prohibiting operations that will cause introduction into the RCS, coolant with a boron concentration less than required to meet the boron concentration of LCO 3.9.1.

This change is acceptable because it applies appropriate controls during periods when RHR is not in operation. The ITS requirement prohibiting operations which would cause a reduction in the RCS boron concentration below that required to maintain the required shutdown margin is necessary to avoid unexpected reactivity changes. Under the ITS definition of CORE ALTERATIONS, many activities which would be considered CORE ALTERATIONS in the CTS, such are core mapping, are not considered CORE ALTERATIONS in the ITS. Therefore, the application of the allowance is expanded in the ITS to cover other activities beyond CORE ALTERATIONS. This change is nominally less restrictive, but represents no practical operational change, and the overall change is considered more restrictive. This change is designated as more restrictive because it imposes a new condition to be met when an RHR loop is not in operation.

M.2 CTS Surveillance 4.9.8.1.2 states that one RHR loop must be verified to be in operation and a. if the RCS temperature is > 140 °F or the time since entry into MODE 3 is < 100 hours, circulating reactor coolant at a flow rate ≥ 3000 gpm, or b. if the RCS temperature is ≤ 140 °F or the time since entry into MODE 3 is ≥ 100 hours, circulating reactor coolant at a flow rate ≥ 2000 gpm. ITS SR 3.9.5.1 requires verification that one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm. This changes the CTS by eliminating the option to reduce RHR flow to 2000 gpm when RCS temperature is ≤ 140 °F or the time since entry into MODE 3 is < 100 hours.

The purpose of Surveillance 4.9.8.1.2 is to ensure that there is sufficient RHR flow for decay heat removal and boron mixing in the RCS. A competing requirement is ensuring that there is sufficient net positive suction head for the RHR pumps to prevent air entrainment and pump cavitation. North Anna license amendment 137 (Unit 1) / 120 (Unit 2) provided a lower RHR flow rate limit for RHR operation when the RCS water level is at mid-loop. This change is acceptable because the lower flow rate is not needed in ITS 3.9.5, RHR - High Water Level, because the reactor water level cannot be at mid-loop while in this specification. Therefore, only the single, higher, RHR flow requirement is needed in ITS 3.9.5. This change is designated as more restrictive because it eliminates a Surveillance acceptance limit which is lower than the remaining limit.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.1 states, in part, that with less than one RHR loop in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. ITS 3.9.5 states that with the RHR loop requirements not met, within 4 hours secure the equipment hatch with at least four bolts, close one door in each installed air lock, and close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent, or verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System. This changes the CTS Actions by allowing penetrations capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System to remain open when the RHR requirements are not met.

The purpose of the CTS 3.9.8.1 Action is to ensure that radioactive material does not escape the containment should the RHR requirements continue to not be met and boiling occurs in the core. Therefore, containment penetrations are closed to seal the containment. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The Required Actions are consistent with the actions taken for containment closure during CORE ALTERATIONS in CTS 3.9.4 and ITS 3.9.4. Penetrations which will be closed by an OPERABLE Containment Purge and Exhaust Isolation system do not need to be closed if RHR is inoperable because the presence of radioactivity in the containment will cause the valves to close automatically, thus performing the isolation function. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.9.8.1.2 states that an RHR loop must be verified to be in operation and providing the required flow at least once per 4 hours. ITS SR 3.9.5.1 requires verification that one RHR loop is operating and providing the required flow every 12 hours. This changes the CTS by reducing the Frequency for performing this Surveillance from 4 to 12 hours.

The purpose of CTS 4.9.8.1.2 is to periodically verify that the RHR system is OPERABLE and operating. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. For CTS 4.9.8.1.2, the Surveillance Frequency of 12 hours is acceptable because there are sufficient indications and alarms available to alert the operator to a malfunction in the RHR system. A once per shift formal verification of operation and flow rate is sufficient to give confidence that the system is operating properly. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.3 *(Category 5 – Deletion of Surveillance Requirement)* CTS Surveillance 4.9.8.1.1 requires verification that each RHR loop is OPERABLE per Specification 4.0.5. ITS 3.9.5 does not contain this Surveillance.

The purpose of CTS Specification 4.0.5 is to require inservice testing in accordance with 10 CFR 50.55a. The purpose of inservice testing of RHR is to detect gross degradation caused by impeller structural damage or other hydraulic component problems. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment 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 function. The Technical Specifications will no longer require RHR to be included in the inservice testing program. This change is acceptable because it is not necessary to perform inservice testing of RHR to determine if it is OPERABLE as the system is routinely operated and the RHR loops are instrumented so that degradation can be observed. Significant degradation of the RHR system would be indicated by the RHR System flow and temperature instrumentation in the Control Room. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.4 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.1 states, in part, that with less than one RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System. ITS 3.9.5, Action A.1, states that with the RHR loop requirements not met, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1. This changes the CTS by allowing coolant with boron concentration less than the RCS boron concentration, but greater than the boron concentration limit in LCO 3.9.1, to be added to the RCS when the RHR requirements are not met.

The purpose of the CTS 3.9.8.1 Action is to ensure that the required shutdown margin is maintained during periods when the RHR requirements are not met. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The Required Actions ensure that the RCS boron concentration is maintained within the limits of LCO 3.9.1, "Refueling Boron Concentration," which is sufficient to ensure that adequate shutdown margin is maintained. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

UNIT 1

ITS

REFUELING OPERATIONS

RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION

LOW WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.6

LCO Notes

3.9.8.2 Two independent RHR loops shall be OPERABLE* with at least one loop in operation. A.2

Insert LCO Note 1 and 2 L.5

APPLICABILITY: MODE 6 with the reactor vessel water level less than 23 feet above the top of the reactor pressure vessel flange.

Action A.1

Insert Proposed Action A.2

Action A.2

Action B.1 & B.2

Insert proposed Actions B.1, B.2, B.3, B.4, B.5.1 and B.5.2

ACTION: a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible. A.3

b. With less than one RHR loop in operation, suspend all operations involving ~~an~~ increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. A.5 L.4 L.1

~~c. The provisions of Specification 3.0.3 are not applicable.~~ A.4

SURVEILLANCE REQUIREMENTS

~~4.9.8.2.1 Verify the required RHR loops to be OPERABLE per Specification 4.0.5.~~ L.3

SR 3.9.6.1

4.9.8.2.2 At least once per ~~4~~ 12 hours, verify at least one RHR Loop is in operation and, L.2

a. if the RCS temperature >140° F or the time since entry into MODE 3 is <100 hours, circulating reactor coolant at a flow rate ≥3000 gpm.

b. if the RCS temperature ≤140° F and the time since entry into MODE 3 is ≥100 hours, circulating reactor coolant at a flow rate ≥2000 gpm to remove decay.

SR 3.9.6.2

Insert Proposed SR 3.9.6.2 M.1

~~* The normal or emergency power source may be inoperable for each RHR loop.~~ A.2

UNIT 2

A.1

ITS 3.9.6

8-27-90

REFUELING OPERATIONS

RESIDUAL HEAT REMOVAL (RHR) AND COOLANT CIRCULATION

LOW WATER LEVELS

LIMITING CONDITION FOR OPERATION

ITS 3.9.6

LCO Notes

3.9.8.2 Two independent RHR loops shall be OPERABLE with at least one loop in operation. A.2

Insert LCO Notes 1 and 2 L.5

APPLICABILITY: MODE 6 with the reactor vessel water level less than 23 feet above the top of the reactor pressure vessel flange.

Action A.1

ACTION: a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

Action A.2 — Insert proposed Action A.2 A.3

Action B.1 + B.2 — Insert Proposed Actions B.1, B.2, B.3, B.4, B.5.1 and B.5.2

b. With less than one RHR loop in operation, suspend all operations involving ~~an increase in the reactor decay heat load or~~ a reduction in boron concentration of the Reactor Coolant System. ~~Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.~~ A.5 L.4 L.1

~~c. The provisions of Specification 3.0.3 are not applicable.~~ A.4

SURVEILLANCE REQUIREMENTS

~~4.9.8.2.1 Verify the required RHR loops to be OPERABLE per Specification 4.0.5.~~ L.3

SR 3.9.6.1

4.9.8.2.2 At least once per 4 hours, verify at least one RHR Loop is in operation and, 12 L.2

a. if the RCS temperature >140° F or the time since entry into MODE 3 is <100 hours, circulating reactor coolant at a flow rate ≥3000 gpm.

b. if the RCS temperature ≤140° F and the time since entry into MODE 3 is ≥100 hours, circulating reactor coolant at a flow rate ≥2000 gpm to remove decay heat.

SR 3.9.6.2 — Insert proposed SR 3.9.6.2 M.1

~~* The normal or emergency power source may be inoperable for each RHR loop.~~ A.2

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 3.9.8.2 LCO is modified by a footnote, *, which states that the normal or emergency power source may be inoperable for each RHR loop. ITS 3.9.6 does not include this statement. The ITS definition of "OPERABLE" states that a component is OPERABLE if either the normal or emergency power source is OPERABLE. This changes CTS by deleting an allowance already provided in a different portion of the ITS.

This change is acceptable because the ITS definition of OPERABLE contains the necessary requirements for a component to perform its safety function. This change is designated as administrative because it does not result in technical changes to the CTS.

A.3 CTS 3.9.8.2, Action a, states, that with less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible. ITS 3.9.6, Condition A, states that with less than the required number of RHR loops OPERABLE, immediately initiate action to restore required RHR loops to OPERABLE status or immediately initiate action to establish ≥ 23 feet of water above the top of reactor vessel flange. This changes the CTS by providing the option to exit the Applicability of the LCO.

This change is acceptable because the requirements have not changed. Exiting the Applicability of LCO is always an option to exit a Condition. Therefore, stating this option explicitly does not change the requirements of the specification. This change is designated as administrative because it does not result in technical changes to the CTS.

A.4 CTS 3.9.8.2 Action c. states, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.6 does not include this statement. ITS LCO 3.0.3 states, "LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4." This changes CTS by deleting an allowance already provided in a different portion of the ITS.

This change is acceptable because ITS LCO 3.0.3 requirements are consistent with those stated in the CTS. This change is designated as administrative because it does not result in technical changes to the CTS.

A.5 CTS 3.9.8.2, Action b, states, in part, that with less than one RHR loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. ITS 3.9.6 states that with no RHR loop in operation, suspend operations involving a reduction in reactor coolant boron concentration. This changes the CTS by eliminating the requirement to suspend operations involving an increase in reactor decay heat load.

This change is acceptable because the requirements have not changed. The reactor decay heat load is generated by irradiated fuel. The only method of increasing the decay head load of a reactor in MODE 6 is to load additional irradiated fuel assemblies into the core. However, ITS LCO 3.9.7 prohibits loading of fuel assemblies into the reactor when the water level is less than 23 feet. Therefore, when LCO 3.9.6 is applicable, there is no method available to increase the reactor decay heat load and the requirement can be deleted with no effect on plant operations. This change is designated as administrative because it does not result in technical changes to the CTS.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.9.8.2 requires two independent RHR loops to be OPERABLE and at least one loop to be in operation. ITS SR 3.9.6.2 requires verification every seven days of correct breaker alignment and that indicated power is available to the RHR pump not in operatoin. A Note states that the Surveillance Requirement is not required to be performed until 24 hours after a required RHR pump is not in operation. This changes the CTS by adding a Surveillance Requirement.

The purpose of ITS 3.9.6 is to require one loop to be in operation and one loop to be held in readiness should it be needed. This change is acceptable because it verifies that the loop that is in standby will be ready should it be needed. This change is designated as more restrictive because it adds a new Surveillance Requirement to the CTS.

RELOCATED SPECIFICATIONS

None

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.2 states, in part, that with less than one RHR loop in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. ITS 3.9.6 states that with no RHR loop in operation, within 4 hours secure the equipment hatch cover with at least four bolts, close one door in each installed air lock, and close each penetration providing direct access from the containment atmosphere to the outside atmosphere with a manual or automatic isolation valve, blind flange, or equivalent, or verify each penetration is capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System. This changes the CTS Actions by allowing penetrations capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System to remain open when no RHR loop is in operation.

The purpose of the CTS 3.9.8.2 Action is to ensure that radioactive material does not escape the containment should the RHR requirements continue to not be met and boiling occurs in the core. Therefore, containment penetrations are closed to seal the containment. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. The Required Actions are consistent with the actions taken for containment closure during CORE ALTERATIONS in CTS 3.9.4 and ITS 3.9.4. Penetrations which will be closed by an OPERABLE Containment Purge and Exhaust Isolation system do not need to be closed if RHR is inoperable because the presence of radioactivity in the containment will cause the valves to close automatically, thus performing the isolation function. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.9.8.2.2 states that an RHR loop must be verified to be in operation and providing the required flow at least once per 4 hours. ITS SR 3.9.6.1 requires verification that one RHR loop is operating and providing the required flow every 12 hours. This changes the CTS by reducing the Frequency for performing this Surveillance from 4 to 12 hours.

The purpose of CTS 4.9.8.2.2 is to periodically verify that the RHR system is OPERABLE and operating. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. For CTS 4.9.8.2.2, the Surveillance Frequency of 12 hours is acceptable because there are sufficient indications and alarms available to alert the operator to a malfunction in the RHR system. Formal verification of operation and flow rate every 12 hours is sufficient to give confidence that the system is operating properly. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

L.3 *(Category 5 – Deletion of Surveillance Requirement)* CTS Surveillance 4.9.8.2.1 requires verification that each RHR loop is OPERABLE per Specification 4.0.5. ITS 3.9.6 does not contain this Surveillance.

The purpose of CTS Specification 4.0.5 is to require inservice testing in accordance with 10 CFR 50.55a. The purpose of inservice testing of RHR is to detect gross degradation caused by impeller structural damage or other hydraulic component problems. This change is acceptable because the deleted Surveillance Requirement is not necessary to verify that the equipment 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 function. The Technical Specifications will no longer require RHR to be included in the inservice testing program. This change is acceptable because it is not necessary to perform inservice testing of RHR to determine if it is OPERABLE as the system is routinely operated and the RHR loops are instrumented so that degradation can be observed. Significant degradation of the RHR system would be indicated by the RHR System flow and temperature instrumentation in the Control Room. This change is designated as less restrictive because Surveillances which are required in the CTS will not be required in the ITS.

L.4 *(Category 4 – Relaxation of Required Action)* CTS 3.9.8.2 states, in part, that with less than one RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System. ITS 3.9.6, Action B.1, states that with the RHR loop requirements not met, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1. This changes the CTS by allowing coolant with boron concentration less than the RCS boron concentration, but greater than the boron concentration limit in LCO 3.9.1, to be added to the RCS when the RHR requirements are not met.

The purpose of the CTS 3.9.8.2 Action is to ensure that the required shutdown margin is maintained during periods when the RHR requirements are not met. This change is acceptable because the Required Actions are used to establish remedial measures that must be taken in response to the degraded conditions in order to minimize risk associated with continued operation while providing time to repair inoperable features. The Required Actions are consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of an accident occurring during the repair period. The Required Actions ensure that the RCS boron concentration is maintained within the limits of LCO 3.9.1, "Refueling Boron Concentration," which is sufficient to ensure that adequate shutdown margin is maintained. This change is designated as less restrictive because less stringent Required Actions are being applied in the ITS than were applied in the CTS.

L.5 *(Category 1 – Relaxation of LCO Requirements)* ITS 3.9.6 is modified by two LCO Notes. Note 1 allows all RHR pumps to be de-energized for ≤ 15 minutes when switching from one train to another, provided several conditions are met. Note 2 allows one required RHR loop to be inoperable for up to 2 hours for Surveillance testing, provided that the other loop is OPERABLE and in operation. CTS 3.9.8.2 does not contain these allowances. This changes the CTS by providing allowing the LCO to not be met.

The purpose of ITS 3.9.6 is to ensure sufficient decay heat removal is available in the specified MODES and conditions. This change is acceptable because the LCO requirements continue to ensure that the structures, systems, and components are maintained consistent with the safety analyses and licensing basis. The ITS Notes allow normal operational evolutions, such as pump swapping and surveillance testing, to be performed while in the applicability of the specification. These evolutions are necessary to demonstrate RHR OPERABILITY. This change is designated as less restrictive because less stringent LCO requirements are being applied in the ITS than were applied in the CTS.

UNIT 1

(A.1)

2-15-89

REFUELING OPERATIONS

WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

ITS

LIMITING CONDITION FOR OPERATION

LCO 3.9.7

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange. (A.3)

APPLICABILITY: ~~MODE 6~~ during movement of irradiated fuel assemblies within the containment. (L.1)

Action A

ACTION: With the requirements of the above specification not satisifed, suspend all operations involving movement of irradiated fuel assemblies within the ~~reactor pressure vessel~~ Containment. ~~The provisions of Specification 3.0.3 are not applicable.~~ (L.1) (M.1) (A.2)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

4.9.10.1 The water level shall be determined to be at least its minimum required depth ~~within 2 hours prior to the start of and~~ at least once per 24 hours ~~thereafter during movement of fuel assemblies~~. (L.2)

ITS 3.9.7

R.1

2-15-89

REFUELING OPERATIONS

WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

APPLICABILITY: MODE 6 during movement of control rods within the reactor pressure vessel.

ACTION: With the requirements of the above specification not satisified, suspend all operations involving movement of control rods within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of control rods within the reactor pressure vessel.

UNIT 2

A.1

ITS 3.9.7

ITS

REFUELING OPERATIONS

WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

LCO 3.9.7

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange. A.3 L1

APPLICABILITY: MODE 6 during movement of irradiated fuel assemblies within the containment.

ACTION: L.1

Action A

With the requirements of the above specification not satisfied, suspend all operations involving movement of irradiated fuel assemblies within the ~~reactor pressure vessel~~ containment. ~~The provisions of Specification 3.0.3 are not applicable.~~ A.2 M.1

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

4.9.10.1 The water level shall be determined to be at least its minimum required depth ~~within 2 hours prior to the start of and~~ at least once per 24 hours ~~thereafter during movement of fuel assemblies.~~ L.2

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I TS 3.9.7

R.1

2-15-89

REFUELING OPERATIONS

WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

APPLICABILITY: MODE 6 during movement of control rods within the reactor pressure vessel.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of control rods within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of control rods within the reactor pressure vessel.

Rev 0

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 3.9.10.1 Action contains the statement, "The provisions of Specification 3.0.3 are not applicable." ITS 3.9.7 does not contain an equivalent statement.

This change is acceptable because the technical requirements have not changed. ITS LCO 3.0.3 is not applicable in MODE 6. Therefore, the LCO 3.0.3 exception is not needed. This change is designates as administrative because the technical requirements of the specifications have not changed.

A.3 CTS 3.9.10.1 is applicable in MODE 6 during movement of fuel assemblies within containment. ITS 3.9.7 is applicable during the movement of irradiated fuel assembles within containment. This changes the CTS by eliminating the "MODE 6" portion of the applicability. Qualification of irradiated fuel vice fuel is discussed in DOC L.1.

This change is acceptable because the technical requirements have not changed. Fuel movement in the containment only occurs in MODE 6. Therefore, specifying MODE 6 during movement of fuel is unnecessary. This change is designates as administrative because the technical requirements of the specifications have not changed.

MORE RESTRICTIVE CHANGES

M.1 CTS 3.9.10.1 Action states that with the reactor vessel water level not within limit, suspend movement of fuel assemblies within the reactor pressure vessel. ITS 3.9.7 states with the refueling cavity water level not within limit, suspend movement of irradiated fuel assemblies within containment. This change the CTS by expanding the suspension of movement of fuel assemblies from within the reactor pressure vessel to within the containment.

The purpose of CTS 3.9.10.1 is to prohibit the occurrence of a fuel handling accident if the refueling cavity water level is less than that assumed in the fuel handling accident analysis. This change is acceptable because the fuel handling accident analysis assumes an irradiated fuel assembly is dropped within the containment, not only within the reactor vessel. In order to protect the initial assumptions of the fuel handling accident analysis, prohibition of irradiated fuel movement within the containment is required. This change is designated as more restrictive because it will prohibit operations that are not prohibited in the CTS.

RELOCATED SPECIFICATIONS

R.1 CTS 3.9.10.2 states that the refueling cavity water level must be at least 23 feet above the fuel during MODE 6 during movement of control rods within the reactor pressure vessel. Movement of control rods is not an initiator of any UFSAR accident analysis. This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because CTS 3.9.10.2 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

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

1. The refueling cavity water level during movement of control rods is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. The refueling cavity water level during movement of control rods does not satisfy criterion 1.

2. The refueling cavity water level during movement of control are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The refueling cavity water level during movement of control rods does not satisfy criterion 2.

3. The refueling cavity water level during movement of control rods is not a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The refueling cavity water level during movement of control rods does not satisfy criterion 3.

4. The refueling cavity water level during movement of control rods is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The refueling cavity water level during movement of control rods was not evaluated in WCAP-11618. However, an evaluation has found that refueling cavity water level during movement of control rods is a non-significant risk contributor to core damage frequency and offsite releases. The refueling

cavity water level during movement of control rods is not important for any scenarios modeled in the North Anna Power Station site-specific PRAs. The refueling cavity water level during movement of control rods does not meet criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the refueling cavity water level during movement of control rods LCO and associated Applicability, Actions, and Surveillances may be relocated out of the Technical Specifications. The refueling cavity water level during movement of control rods specification will be relocated to the TRM. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the LCO did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

REMOVED DETAIL CHANGES

None

LESS RESTRICTIVE CHANGES

L.1 *(Category 2 – Relaxation of Applicability)* CTS 3.9.10.1 states that at least 23 feet of water must be maintained over the reactor pressure vessel flange in MODE 6 during movement of fuel assemblies within the containment. The 3.9.10.1 Action requires suspension of movement of fuel assemblies if the water level requirement is not met. ITS 3.9.7 states the refueling cavity water level shall be maintained ≥ 23 feet above the top of the reactor vessel flange during movement of irradiated fuel assemblies within containment. ITS 3.9.7, Action A.2, requires the suspension of movement o irradiated fuel assemblies within containment. This changes the CTS restricting the applicability and Actions from movement of any fuel assemblies within containment to movement of irradiated fuel within containment. The change eliminating MODE 6 is discussed in DOC A.3.

The purpose of CTS 3.9.10.1 is to prohibit the occurrence of a fuel handling accident if the refueling cavity water level is less than that assumed in the fuel handling accident analysis. This change is acceptable because the requirements continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The fuel handling accident analysis is based on the dropping of an irradiated fuel assembly. An unirradiated fuel assembly does not contain the radioactive materials generated by fission and does not result in significant offsite doses if dropped. This change is designated as less restrictive because the LCO requirements are applicable in fewer operating conditions than in the CTS.

L.2 *(Category 7 – Relaxation Of Surveillance Frequency)* CTS 4.9.10.1 requires the refueling cavity water level to be determined to be within limit within 2 hours prior to the start of and at least once per 24 hours during movement of fuel assemblies. ITS SR 3.7.9.1 requires verification that the refueling cavity water level is within limit every 24 hours. This changes the CTS by reducing the Frequency for verifying refueling cavity water level from 2 hours before fuel movement to 24 hours before fuel movement.

The purpose of CTS 4.9.10.1 is to prohibit the movement of fuel, and thereby prohibit a fuel handling accident, if the refueling cavity water level is less than that assumed in the fuel handling accident analysis. This change is acceptable because the new Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. The Frequency of 24 hours is sufficient during the movement of fuel and is, therefore, sufficient before fuel is moved. ITS SR 3.0.1 requires the SR to be met during the MODES or other specified conditions in the Applicability. Therefore, the water level must be met when fuel is moved or fuel movement must be suspended immediately (thereby exiting the applicability of the specification). Therefore, changing the Frequency from 2 hours before moving fuel to within 24 hours before moving fuel has no effect on plant safety. This change is designated as less restrictive because Surveillances will be performed less frequently under the ITS than under the CTS.

UNIT 1

CTS 3.9.3

8-21-85

~~REFUELING OPERATIONS~~

~~DECAY TIME~~

~~**LIMITING CONDITION FOR OPERATION**~~

~~3.9.3 The reactor shall be subcritical for at least 150 hours.~~

~~APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.~~

~~ACTION:~~

~~With the reactor subcritical for less than 150 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.~~

R.1

~~**SURVEILLANCE REQUIREMENTS**~~

~~4.9.3 The reactor shall be determined to have been subcritical for at least 150 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.~~

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UNIT 2

CTS 3.9.3

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 150 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 150 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 150 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

R.1

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RELOCATED SPECIFICATIONS

R.1 CTS 3.9.3 states that the reactor must be subcritical for at least 150 hours prior to movement of movement of irradiated fuel in the reactor pressure vessel. This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

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

1. Decay time is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Decay time does not satisfy criterion 1.

2. Decay time is a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Specifically, the fuel handling accident analysis assumes 100 hours of decay time before fuel movement. However, the 100 hour decay time following subcriticality will always be met for a refueling outage because of the operations required prior to moving irradiated fuel in the reactor vessel (e.g., containment entry, removal of the reactor vessel head, and removal of the reactor vessel internals). Therefore, this requirement is not a limiting condition for operation and decay time does not satisfy criterion 2.

3. The decay time limit is not a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Decay time does not satisfy criterion 3.

4. Decay time is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. Decay time was not evaluated in WCAP-11618. However, an evaluation has found decay time is a non-significant risk contributor to core damage frequency and offsite releases. Decay time is not important for any scenarios modeled in the North Anna Power Station site-specific PRAs. Decay time does not meet criterion 4.

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

UNIT 1

CTS 3.9.5

11-26-77

~~REFUELING OPERATIONS~~

~~COMMUNICATIONS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.~~

~~APPLICABILITY: During CORE ALTERATIONS.~~

~~ACTION:~~

~~When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS. Written documentation of the 12 hour checks is not required.~~

R.1

Rev 0

UNIT 2

CTS 3.9.5

8-21-80

REFUELING OPERATIONS

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

R.1

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS. Written documentation of the 12 hour checks is not required.

RELOCATED SPECIFICATIONS

R.1 CTS 3.9.5 states that direct communications shall be maintained between the control room and personnel at the refueling station during CORE ALTERATIONS. This ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS. The prompt notification of the control room of a fuel handling accident is an assumption in the Fuel Handling Analysis. This prompt notification is used to ensure that the control room is isolated promptly and is necessary to meet the control room operator dose limits in General Design Criteria 19. This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

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

1. Communications are not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Communications does not satisfy criterion 1.

2. Communications are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Communications does not satisfy criterion 2.

3. Communications is part of the primary success path and is assumed in the mitigation of a DBA which assumes the failure of a fission product barrier. However, communications is not a structure, system or component. Communications does not satisfy criterion 3.

4. Communications is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. As discussed in Section 4.0, (Appendix A, page A-67) of WCAP-11618, communications was found to be a non-significant risk contributor to core damage frequency and offsite releases. The Company has reviewed this evaluation, considers it applicable to the North Anna Power Station, and concurs with this assessment. Communications is not important for any scenarios modeled in the North Anna Power Station site-specific PRAs. Communications do not meet criterion 4.

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

UNIT 1

CTS 3.9.6

REFUELING OPERATIONS

MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1. A minimum capacity of 3250 pounds, and
2. An overload cut off limit $\leq$ 2850 pounds.

b. The auxiliary hoist used for movement of control rods having:

1. A minimum capacity of 700 pounds, and
2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

R.1

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2850 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.

UNIT 2

CTS 3.9.6

8-21-80

REFUELING OPERATIONS

MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1. A minimum capacity of 3250 pounds, and
2. An overload cut off limit less than or equal to 2850 pounds.

b. The auxiliary hoist used for movement of control rods having:

1. A minimum capacity of 700 pounds, and
2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

R.1

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2850 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.

RELOCATED SPECIFICATIONS

R.1 CTS 3.9.6 states that the manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE during movement of control rods or fuel assemblies within the reactor pressure vessel. This specification ensures that the lifting device on the Manipulator Crane has adequate capacity to lift the weight of a fuel assembly and a Rod Control Cluster Assembly, and that an automatic load limiting device is available to prevent damage to the fuel assembly during fuel movement. This specification also ensures that the auxiliary hoist on the Manipulator Crane has adequate capacity for latching and unlatching control rod drive shafts. This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

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

1. Manipulator Crane OPERABILITY is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary Manipulator Crane OPERABILITY does not satisfy criterion 1.

2. Manipulator Crane OPERABILITY is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Manipulator Crane OPERABILITY does not satisfy criterion 2.

3. Manipulator Crane OPERABILITY is not a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Manipulator Crane OPERABILITY does not satisfy criterion 3.

4. Manipulator Crane OPERABILITY is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. As discussed in Section 4.0, (Appendix A, page A-68) of WCAP-11618, Manipulator Crane OPERABILITY was found to be a non-significant risk contributor to core damage frequency and offsite releases. The Company has reviewed this evaluation, considers it applicable to the North Anna Power Station, and concurs with this assessment. Manipulator Crane OPERABILITY is not important for any scenarios modeled in the North Anna Power Station site-specific PRAs. Manipulator Crane OPERABILITY does not meet criterion 4.

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

UNIT 1

CTS 3.9.7

R.1

REFUELING OPERATIONS

CRANE TRAVEL – SPENT FUEL PIT

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2500 pounds shall be prohibited from travel over irradiated fuel assemblies in the spent fuel pit. This does not apply to movement of any spent fuel pit gate provided each of the following is satisfied:

a. the top of the gate (excluding lifting lugs) is no higher than 15 inches above the top of the moveable platform crane deck support beam while over irradiated fuel,

b. the gate is rigged to slack-free safety cables while over irradiated fuel,

c. irradiated fuel containing Rod Control Cluster Assemblies are excluded along the load path where the gate is moved, and

d. irradiated fuel is prohibited in the cask area when the gate is lifted over the spent fuel cask handling area. There is no restriction on lift height.

APPLICABILITY: With irradiated fuel assemblies in the spent fuel pit.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Loads other than the spent fuel pit gates shall be verified to be less than 2500 pounds prior to movement over irradiated fuel assemblies in the spent fuel pit.

4.9.7.2 For movement of any of the spent fuel pit gates:

a. gate lift height and slack-free redundant rigging shall be verified prior to moving over irradiated fuel,

b. load paths shall be verified not to have irradiated fuel with Rod Control Cluster Assemblies present in the gate load path, and

c. the spent fuel cask handling area shall be verified to have no irradiated fuel present prior to moving a gate over the area.

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UNIT 2

CTS 3.9.7

REFUELING OPERATIONS

CRANE TRAVEL - SPENT FUEL PIT

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2500 pounds shall be prohibited from travel over irradiated fuel assemblies in the spent fuel pit. This does not apply to movement of any spent fuel pit gate provided each of the following is satisfied:

a. the top of the gate (excluding lifting lugs) is no higher than 15 inches above the top of the moveable platform crane deck support beam while over irradiated fuel,

b. the gate is rigged to slack-free safety cables while over irradiated fuel,

c. irradiated fuel containing Rod Control Cluster Assemblies are excluded along the load path where the gate is moved, and

d. irradiated fuel is prohibited in the cask area when the gate is lifted over the spent fuel cask handling area. There is no restriction on lift height.

R.1

APPLICABILITY: With irradiated fuel assemblies in the spent fuel pit.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7.1 Loads other than the spent fuel pit gates shall be verified to be less than 2500 pounds prior to movement over irradiated fuel assemblies in the spent fuel pit.

4.9.7.2 For movement of any of the spent fuel pit gates:

a. gate lift height and slack-free redundant rigging shall be verified prior to moving over irradiated fuel,

b. load paths shall be verified not to have irradiated fuel with Rod Control Cluster Assemblies present in the gate load path, and

c. the spent fuel cask handling area shall be verified to have no irradiated fuel present prior to moving a gate over the area.

RELOCATED SPECIFICATIONS

R.1 CTS 3.9.7 places restriction on movement of loads over irradiated assemblies in the spent fuel pit in excess of 2500 pounds. This represents the working load of the fuel assembly plus gripper. The LCO ensures that in the event this load is dropped the activity release will be limited to that contained in a single fuel assembly and any possible distortion of fuel in the storage racks will not result in a critical array. This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

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

1. Crane Travel - Spent Fuel Pit is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Crane Travel - Spent Fuel Pit does not satisfy criterion 1.

2. Crane Travel - Spent Fuel Pit is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Crane Travel - Spent Fuel Pit does not satisfy criterion 2.

3. Crane Travel - Spent Fuel Pit is not a structure, system or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Crane Travel - Spent Fuel Pit does not satisfy criterion 3.

4. Crane Travel - Spent Fuel Pit is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. As discussed in Section 4.0, (Appendix A, page A-68) of WCAP-11618, Crane Travel - Spent Fuel Pit was found to be a non-significant risk contributor to core damage frequency and offsite releases. The Company has reviewed this evaluation, considers it applicable to the North Anna Power Station, and concurs with this assessment. Crane Travel - Spent Fuel Pit is not important for any scenarios modeled in the North Anna Power Station site-specific PRAs. Crane Travel - Spent Fuel Pit does not meet criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the Crane Travel - Spent Fuel Pit LCO and associated Applicability, Actions, and Surveillances may be relocated out of the Technical Specifications. The Crane Travel - Spent Fuel Pit specification will be relocated to the TRM which is incorporated by reference into the UFSAR. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

This change is designated as relocation because the LCO did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and has been relocated to the TRM.

UNIT 1

CTS 3.9.9

11-26-77

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provision of Specification 3.0.3 are not applicable.

R.1

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from the containment gaseous and particulate radiation monitoring instrumentation channels.

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UNIT 2

CTS 3.9.9

8-21-80

REFUELING OPERATIONS

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Purge and Exhaust isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the Containment Purge and Exhaust isolation system inoperable, close each of the Purge and Exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere.
The provisions of Specification 3.0.3 are not applicable.

R.1

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Purge and Exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment Purge and Exhaust isolation occurs on manual initiation and on a high radiation test signal from the containment gaseous and particulate radiation monitoring instrumentation channels.

RELOCATED SPECIFICATIONS

R.1 CTS 3.9.9 states requirements for the containment purge and exhaust isolation system, which automatically closes the containment purge and exhaust isolation valves in MODE 6. This LCO does not meet the criteria for retention in the ITS; therefore, it will be retained in the Technical Requirements Manual.

This change is acceptable because 3.9.9 does not meet the 10 CFR 50.36(c)(2)(ii) criteria for inclusion into the ITS.

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

1. The containment purge and exhaust isolation system is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. The containment purge and exhaust isolation system does not satisfy criterion 1.

2. The containment purge and exhaust isolation system is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or Transient Analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The containment purge and exhaust isolation system does not satisfy criterion 2.

3. The containment purge and exhaust isolation system is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The containment purge and exhaust valves are not assumed to close in case of a Fuel Handling Accident inside containment. The containment purge and exhaust isolation system does not satisfy criterion 3.

4. The containment purge and exhaust isolation system is not a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The Company has reviewed this evaluation, considers it applicable to the North Anna Power Station, and concurs with this assessment. The containment purge and exhaust isolation system is not important for any scenarios modeled in the North Anna Power Station site-specific PRAs. The containment purge and exhaust isolation area system does not meet criterion 4.

Since the 10 CFR 50.36(c)(2)(ii) criteria have not been met, the containment purge and exhaust isolation system requirements and associated Applicability, Actions, and Surveillances may be relocated out of the Technical Specifications. The containment purge and exhaust isolation system requirements will be relocated to the TRM which is incorporated by reference into the UFSAR. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. This change is designated as relocation because the requirements did not meet the criteria in 10 CFR 50.36(c)(2)(ii) and have been relocated to the TRM.

SECTION 3.9 - REFUELING OPERATIONS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

GENERIC NSHCs

10 CFR 50.92 EVALUATION
FOR
ADMINISTRATIVE CHANGES

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications with no change in intent. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording that does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1431. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not affect initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR MORE RESTRICTIVE CHANGES

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements that currently do not exist.

These changes include additional commitments that decrease allowed outage times, increase the frequency of surveillances, impose additional surveillances, increase the scope of specifications to include additional plant equipment, increase the applicability of specifications, or provide additional actions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The imposition of more restrictive requirements either has no effect on or increases the margin of plant safety. As provided in the discussion of change, each change in this category is, by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR RELOCATED SPECIFICATIONS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relocating existing Technical Specification LCOs to licensee controlled documents.

The the Company has evaluated the current Technical Specifications using the criteria set forth in 10 CFR 50.36. Specifications identified by this evaluation that did not meet the retention requirements specified in the regulation are not included in the Improved Technical Specifications (ITS) submittal. These specifications have been relocated from the current Technical Specifications to the Technical Requirements Manual.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria of 10 CFR 50.36 (c)(2)(ii) for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the North Anna Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to the Technical Requirements Manual, which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR.50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no significant effect on any safety analyses assumptions, as indicated by the fact that the requirements do not meet the 10 CFR 50.36 criteria for retention: In addition, the relocated requirements are moved without change and any future changes to these requirements will be evaluated per 10 CFR 50.59.

NRC prior review and approval of changes to these relocated requirements, in accordance with 10 CFR 50.92, will no longer be required. This review and approval does not provide a specific margin of safety which can be evaluated. However, since the proposed change is consistent with the Westinghouse Standard Technical Specifications, NUREG-1431 issued by the NRC, revising the Technical Specifications to reflect the approved level of detail gives assurance that this relocation does not result in a significant reduction in the margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES - REMOVED DETAIL

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve moving details out of the Technical Specifications and into the Technical Specifications Bases, the UFSAR, the TRM or other documents under regulatory control such as the Quality Assurance Program Topical Report. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1431 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relocates certain details from the Technical Specifications to other documents under regulatory control. The Bases, UFSAR, and Technical Requirement Manual will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the Technical Specifications. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e). Other documents are subject to controls imposed by Technical Specifications or regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no effect on any safety analysis assumptions. In addition, the details to be moved from the Technical Specifications to other documents are not being changed. Since any future changes to these details will be evaluated under the applicable regulatory change control mechanism, no significant reduction in a margin of safety will be allowed. A significant reduction in the margin of safety is not associated with the elimination of the 10 CFR 50.92 requirement for NRC review and approval of future changes to the relocated details. The proposed change is consistent with the Westinghouse Standard Technical Specifications, NUREG-1431, issued by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail, which indicates that there is no significant reduction in the margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 1
RELAXATION OF LCO REQUIREMENTS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by the elimination of specific items from the LCO or Tables referenced in the LCO, or the addition of exceptions to the LCO.

These changes reflect the ISTS approach to provide LCO requirements that specify the protective conditions that are required to meet safety analysis assumptions for required features. These conditions replace the lists of specific devices used in the CTS to describe the requirements needed to meet the safety analysis assumptions. The ITS also includes LCO Notes which allow exceptions to the LCO for the performance of testing or other operational needs. The ITS provides the protection required by the safety analysis and provides flexibility for meeting the conditions without adversely affecting operations since equivalent features are required to be OPERABLE. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change provides less restrictive LCO requirements for operation of the facility. These less restrictive LCO requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the current safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the change is consistent with the assumptions in the current safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The imposition of less restrictive LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the current safety analyses and licensing basis requirements are maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 2 RELAXATION OF APPLICABILITY

The North Anna Nuclear Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the applicability of current Technical Specification (CTS) Limiting Conditions for Operation (LCOs) by reducing the conditions under which the LCO requirements must be met.

Reactor operating conditions are used in CTS to define when the LCO features are required to be OPERABLE. CTS Applicabilities can be specific defined terms of reactor conditions or more general such as, "all MODES" or "any operating MODE." Generalized applicability conditions are not contained in ITS, therefore the ITS eliminates CTS requirements such as "all MODES" or "any operating MODE," replacing them with ITS defined MODES or applicable conditions that are consistent with the application of the plant safety analysis assumptions for operability of the required features.

CTS requirements may also be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or which are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the TS may be satisfied by exiting the applicability which takes the plant out of the conditions that require the safety system to be OPERABLE.

This change provides the protection required by the safety analysis and provides flexibility for meeting limits by restricting the application of the limits to the conditions assumed in the safety analyses. The ITS is also consistent with the plant current licensing basis, as may be modified in the discussion of individual changes. The change is generally made to conform with NUREG-1431 and has been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relaxes the conditions under which the LCO requirements for operation of the facility must be met. These less restrictive applicability requirements for the LCOs do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event in that the requirements continue to ensure that process variables, structures, systems, and components are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does impose different requirements. However, the requirements are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed applicability of LCO requirements does not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to ensure that the LCO requirements are applied in the MODES and specified conditions assumed in the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 3 RELAXATION OF COMPLETION TIME

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the Completion Times for Required Actions in the current Technical Specifications (CTS).

Upon discovery of a failure to meet an LCO, the ITS specifies times for completing Required Actions of the associated TS Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken within specified Completion Times (referred to as Allowed Outage Times (AOTs) in the CTS). These times define limits during which operation in a degraded condition is permitted. Adopting Completion Times from the ITS is acceptable because the Completion Times take into account the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. In addition, the ITS provides consistent Completion Times for similar conditions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change relaxes the Completion Time for a Required Action. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Completion Time does not significantly increase the probability of any accident previously evaluated. The consequences of an analyzed accident during the relaxed Completion Time are the same as the consequences during the existing AOT. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the method governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed Completion Time for a Required Action does not involve a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the allowed Completion Time is consistent with safe operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 4
RELAXATION OF REQUIRED ACTION

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve relaxation of the Required Actions in the current Technical Specifications (CTS).

Upon discovery of a failure to meet an LCO, the ITS specifies Required Actions to complete for the associated Conditions. Required Actions of the associated Conditions are used to establish remedial measures that must be taken in response to the degraded conditions. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from the ISTS is acceptable because the Required Actions take into account the operability status of redundant systems of required features, the capacity and capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relaxes Required Actions. Required Actions and their associated Completion Times are not initiating conditions for any accident previously evaluated and the accident analyses do not assume that required equipment is out of service prior to the analyzed event. Consequently, the relaxed Required Actions do not significantly increase the probability of any accident previously evaluated. The Required Actions in the ITS have been developed to provide appropriate remedial actions to be taken in response to the degraded condition considering the operability status of the redundant systems of required features, and the capacity and capability of remaining features while minimizing the risk associated with continued operation. As a result, the consequences of any accident previously evaluated are not significantly increased. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The Required Actions and associated Completion Times in the ITS have been evaluated to ensure that no new accident initiators are introduced. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed Required Actions do not involve a significant reduction in the margin of safety. As provided in the discussion of change, this change has been evaluated to minimize the risk of continued operation under the specified Condition, considering the operability status of the redundant systems of required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, and the low probability of a DBA occurring during the repair period. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 5
DELETION OF SURVEILLANCE REQUIREMENT

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve deletion of Surveillance Requirements in the current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. The ITS eliminates unnecessary CTS Surveillance Requirements that do not contribute to verification that the equipment 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. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change deletes Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The remaining Surveillance Requirements are consistent with industry practice and are considered to be sufficient to prevent the removal of the subject Surveillances from creating a new or different type of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deleted Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the change has been evaluated to ensure that the deleted Surveillance Requirements are not necessary for verification that the equipment 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. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 6
RELAXATION OF SURVEILLANCE REQUIREMENT ACCEPTANCE CRITERIA

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the relaxation of Surveillance Requirements acceptance criteria in the current Technical Specifications (CTS).

The CTS require safety systems to be tested and verified Operable prior to entering applicable operating conditions. The ITS eliminates or relaxes the Surveillance Requirement acceptance criteria that do not contribute to verification that the equipment used to meet the LCO can perform its required functions. For example, the ITS allows some Surveillance Requirements to verify Operability under actual or test conditions. Adopting the ITS allowance for "actual" conditions is acceptable because required features cannot distinguish between an "actual" signal or a "test" signal. Also included are changes to CTS requirements that are replaced in the ITS with separate and distinct testing requirements which, when combined, include Operability verification of all TS required components for the features specified in the CTS. Adopting this format preference in the ISTS is acceptable because Surveillance Requirements that remain include testing of all previous features required to be verified OPERABLE. Changes which provide exceptions to Surveillance Requirements to provide for variations which do not affect the results of the test are also included in this category. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relaxes the acceptance criteria of Surveillance Requirements. Surveillances are not initiators to any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased. The equipment being tested is still required to be Operable and capable of performing the accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The relaxed acceptance criteria for Surveillance Requirements do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxed Surveillance Requirement acceptance criteria have been evaluated to ensure that they are sufficient to verify that the equipment used to meet the LCO can perform its required functions. Thus, appropriate equipment continues to be tested in a manner that gives confidence that the equipment can perform its assumed safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION FOR LESS RESTRICTIVE CHANGES – CATEGORY 7 RELAXATION OF SURVEILLANCE FREQUENCY

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the relaxation of Surveillance Frequencies in the current Technical Specifications (CTS).

CTS and ITS Surveillance Frequencies specify time interval requirements for performing surveillance testing. Increasing the time interval between Surveillance tests in the ITS results in decreased equipment unavailability due to testing which also increases equipment availability. In general, the ITS contain test frequencies that are consistent with industry practice or industry standards for achieving acceptable levels of equipment reliability. Adopting testing practices specified in the ITS is acceptable based on similar design, like-component testing for the system application and the availability of other Technical Specification requirements which provide regular checks to ensure limits are met. Relaxation of Surveillance Frequency can also include the addition of Surveillance Notes which allow testing to be delayed until appropriate unit conditions for the test are established, or exempt testing in certain MODES or specified conditions in which the testing can not be performed.

Reduced testing can result in a safety enhancement because the unavailability due to testing is reduced and; in turn, reliability of the affected structure, system or component should remain constant or increase. Reduced testing is acceptable where operating experience, industry practice or the industry standards such as manufacturers' recommendations have shown that these components usually pass the Surveillance when performed at the specified interval, thus the frequency is acceptable from a reliability standpoint. Surveillance Frequency changes to incorporate alternate train testing have been shown to be acceptable where other qualitative or quantitative test requirements are required which are established predictors of system performance. Surveillance Frequency extensions can be based on NRC-approved topical reports. The NRC staff has accepted topical report analyses that bound the plant-specific design and component reliability assumptions. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relaxes Surveillance Frequencies. The relaxed Surveillance Frequencies have been established based on achieving acceptable levels of equipment reliability. Consequently, equipment which could initiate an accident previously evaluated will continue to operate as expected and the probability of the initiation of any accident previously evaluated will not be significantly increased. The equipment being tested is still required to be Operable and capable of performing any accident mitigation functions assumed in the accident analysis. As a result, the consequences of any accident previously evaluated are not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The relaxed Surveillance Frequencies do not result in a significant reduction in the margin of safety. As provided in the discussion of change, the relaxation in the Surveillance Frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a Frequency that gives confidence that the equipment can perform its assumed safety function when required. Therefore, this change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
FOR
LESS RESTRICTIVE CHANGES – CATEGORY 8
DELETION OF REPORTING REQUIREMENTS

The North Anna Power Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants." Some of the proposed changes involve the deletion of requirements in the current Technical Specifications (CTS) to send reports to the NRC.

The CTS includes requirements to submit reports to the NRC under certain circumstances. However, the ITS eliminates these requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The ITS changes to reporting requirements are acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change has no effect on the safe operation of the plant. These changes are generally made to conform with NUREG-1431 and have been evaluated to not be detrimental to plant safety.

In accordance with the criteria set forth in 10 CFR 50.92, the Company has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change deletes reporting requirements. Sending reports to the NRC is not an initiator to any accident previously evaluated. Consequently, the probability of any accident previously evaluated is not significantly increased. Sending reports to the NRC has no effect on the ability of equipment to mitigate an accident previously evaluated. As a result, the consequences of any accident previously evaluated is not significantly affected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The deletion of reporting requirements does not result in a significant reduction in the margin of safety. The ITS eliminates the requirements for many such reports and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The change to reporting requirements does not affect the margin of safety because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change does not involve a significant reduction in a margin of safety.

This proposed Technical Specification change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The following is a discussion of how the proposed Technical Specification change meets the criteria for categorical exclusion.

10 CFR 51.22(c)(9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements,

(i) proposed change involves No Significant Hazards Considerations (refer to the Determination of No Significant Hazards Considerations section of this Technical Specification Change Request);

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths; and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental affect statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed change of this request.

SECTION 3.9 - REFUELING OPERATIONS

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

SPECIFIC NSHCs

There are no specific NSHC discussions for this Section.