ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	C.1 Restore channel or train to OPERABLE status.	48 hours
	OR	
	C.2 Open RTBs.	49 hours
D. One Power Range Neutron Flux—High channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. ----------------------------	
	D.1.1 Place channel in trip.	6 hours
	AND	
	D.1.2 Reduce THERMAL POWER to $\leq$ 75% RTP.	12 hours
	OR	
	D.2.1 Place channel in trip.	6 hours
	AND	
	------------NOTE------------ Only required to be performed when the Power Range Neutron Flux input to QPTR is inoperable. ----------------------------	
	D.2.2 Perform SR 3.2.4.2.	Once per 12 hours
	OR	
	D.3 Be in MODE 3.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.7	------------------NOTE------------------ Not required to be performed for Source Range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. -- Perform COT.	92 days (31 days for Function 5)
SR 3.3.1.8	------------------NOTE------------------ This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -- Perform COT.	------NOTE----- Only required when not performed within previous 92 days (31 days for Functions 4 and 5) -------------- Prior to reactor startup <u>AND</u> Four hours after reducing power below P-10 for power and intermediate range instrumentation <u>AND</u>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8 (continued)		Four hours after reducing power below P-6 for source range instrumentation AND Every 92 days thereafter (31 days for Functions 4 and 5)
SR 3.3.1.9	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	92 days
SR 3.3.1.10	-------------------NOTE------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV. The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

The protection Functions associated with the EAGLE-21™ Process Protection System have an installed bypass capability, and may be tested in either the trip or bypass mode, as approved in Reference 7. When testing is performed in the bypass mode, the SSPS input relays are not operated, as justified in Reference 9. The input relays are checked during the CHANNEL CALIBRATION every 18 months.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days except for source range instrument channels which are every 31 days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of References 6 and 7.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for greater than 4 hours, this Surveillance must be performed within 4 hours after entry into MODE 3.

The Frequency of 92 days is justified in Reference 7, except for Function 13. The justification for Function 13 is provided in Reference 9.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days (31 days for source and intermediate range instrument channels) prior to reactor startup and 4 hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.8 (continued)

performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "4 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days (31 days for source and intermediate range instrument channels) thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range-low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 92 days, as justified in Reference 7.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 (continued)

SR 3.3.1.15 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. Watts Bar FSAR, Section 6.0, "Engineered Safety Features."

2. Watts Bar FSAR, Section 7.0, "Instrumentation and Controls."

3. Watts Bar FSAR, Section 15.0, "Accident Analysis."

4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.

5. 10 CFR Part 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."

6. WCAP-12096, Rev. 6, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," May 1994.

7. WCAP-10271-P-A, Supplement 1, and Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," May 1986 and June 1990.

8. Watts Bar Technical Requirements Manual, Section 3.3.1, "Reactor Trip System Response Times."

9. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar.

10. ISA-DS-67.04, 1982, "Setpoint for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants."

BASES

SURVEILLANCE REQUIREMENTS (continued)

The protection Functions associated with the EAGLE-21™ Process Protection System have an installed bypass capability, and may be tested in either the trip or bypass mode, as approved in Reference 7. When testing is performend in the bypass mode, the SSPS input relays are not operated, as justified in Reference 10. The input relays are checked during the CHANNEL CALIBRATION every 18 months.

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible

(continued)

BASES

REFERENCES (continued)

6. WCAP-12096, Rev. 6, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," May 1994.

7. WCAP-10271-P-A, Supplement 1 and Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System." May 1986 and June 1990.

8. Watts Bar Technical Requirements Manual, Section 3.3.2, "Engineered Safety Feature Response Times."

9. TVA Letter to NRC, November 9, 1984, "Request for Exemption of Quarterly Slave Relay Testing, (L44 841109 808)."

10. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar.

11. Westinghouse letter to TVA (WAT-D-8347), September 25, 1990, "Charging/Letdown Isolation Transients" (T33 911231 810).

Disposition of NRC Technical Specification Open Item

TAC M76742-12

Open item - Page B 3.3-65, LCO 3.3.3, Background Field Transmitters or Sensors
- See Item 10 above.

Disposition

This item is addressed under TAC Item TAC M76742-10.

Disposition of NRC Technical Specification Open Item

TAC M76742-13

Open item - Page B 3.3-95, LCO 3.3.2, Applicable Safety Analysis, Item 7.b - The description and safety analysis for this function needs to be revised to conform to the Watts Bar design.

Disposition

The attached markup of the Westinghouse Technical Specification certification pages address this item. The revised Technical Specification Bases pages are included in Enclosure 4.

Revised Pg B 3.3-91

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

W

2-5-95

7. Automatic Switchover to Containment Sump (continued)

Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

a. Automatic Switchover to Containment Sump-Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Low Coincident With Safety Injection and Coincident With Containment Sump Level-High

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low low level in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability.

The RWST—Low Low ~~Allowable Value~~ Trip Setpoint ~~has both upper and lower limits. The lower limit~~ is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Low Coincident With Safety Injection and Coincident With Containment Sump Level-High (continued)

This setpoint will also ~~The upper limit is selected to~~ ensure that enough borated water is injected to ~~ensure~~ maintain the reactor ~~remains~~ shut down. The ~~high~~ limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction.

W

2-5-95

The transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Automatic switchover occurs only if the RWST low low level signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. The automatic switchover Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

Additional protection from spurious switchover is provided by requiring a Containment Sump Level-High signal as well as RWST Level-Low Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level-High signal must be

(continued)

Disposition of NRC Technical Specification Open Item

TAC M76742-14

Open item - Page B 3.3-120, LCO 3.3.2, References -The brackets around Reference 10 need to be removed or reference deleted due to nonapplicability to Watts Bar.

Disposition

This item is addressed under TAC Item TAC M76742-11.

Disposition of NRC Technical Specification Open Item

TAC M76742-15

Open item - Page B 3.3-130, LCO 3.3.3, LCO Section, Function 11 - The proposed bases change is dependent on resolution of Item 3 above.

Disposition

This item is addressed under TAC Item TAC M76742-3.

Disposition of NRC Technical Specification Open Item

TAC M76742-16

Open item - Page B 3.3-132, LCO 3.3.3, LCO Section, Functions 18 through 21 - The description and safety analysis for the core exit thermocouples needs to be revised to show how Watts Bar meets the intent of II.F.2 of NUREG-0737.

Disposition

The attached markups indicate the changes necessary to address this item. The revised Technical Specification pages are included in Enclosure 4.

BASES

LCO (continued)

17. AFW Valve Status

The status of each AFW swapover to Essential Raw Cooling Water (ERCW) valve is monitored with non-Type A Category 1 indication in the control room. Indication on each valve for fully open or fully closed position is provided. AFW valve status is monitored to give verification to the operator that automatic transfer to ERCW has taken place.

18, 19, 20, 21. Core Exit Temperature

INSERT A

~~Core Exit Temperature is provided for verification and long term surveillance of core cooling.~~

~~The ICCM is used to monitor the core exit thermocouples. There are two isolated systems, with each system monitoring at least four thermocouples per quadrant. The plasma display gives the average quadrant value, the high value, and the low value.~~

~~Control room indications are provided through the ICCM plasma display, which is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with this portion of the instrument channel. Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.~~

~~Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core.~~

(continued)

INSERT A

18, 19, 20, 21. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

Core exit thermocouples, in conjunction with RCS wide range temperatures, are sufficient to provide indication of radial distribution of the coolant entalpy rise across representative sections of the core. Core Exit Temperature is used to support BA 1/27/95 determine BA 1/27/95 whether which of to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabalization and cooldown control. MEB 1/27/95 95-004

The Inadequate Core Cooling Monitor (ICCM) is used to monitor the core exit thermocouples. There are two isolated systems, with each system monitoring at least four thermocouples ~~(channels)~~ per quadrant. The plasma display gives the average quadrant value, the high quadrant value, and the low quadrant value for each quadrant. 95-004

1/27/95

Two OPERABLE channels are required in each quadrant to provide adequate indication of coolant temperature rise in representative regions of the core. Two isolated channels of two thermocouples each ensure a single failure will not disable the ability to identify significant temperature gradients. 95-004

The Watts Bar incore thermocouple monitoring system ~~A~~ supports the plant operating procedures described in Reference 4.

SL 1/27/95

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.4

SR 3.3.3.4 is the performance of a TADOT. This test is performed every 18 months. The test checks operation of the containment isolation valve position indicators and AFW valve position indicators. The Frequency is based on the known reliability of the indicators and has been shown to be acceptable through operating experience.

This SR has been modified by two Notes. Note 1 excludes verification of setpoints for the valve position indicators. Note 2 indicates that this SR is only applicable to Functions 11 and 17, which are the only Functions with valve position indicators.

REFERENCES

1. NUREG-0847, Safety Evaluation Report, Supplement Number 9, June 16, 1992, Section 7.5.2, "Post Accident Monitoring System."

2. Regulatory Guide 1.97, Revision 2, December 1980, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

3. NUREG-0737, "Clarification of TMI Action Plan Requirements," Supplement 1, January 1983.

4. Submittal From John H. Garrity to U.S. Nuclear Regulatory Commission dated January 24, 1992, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - NUREG 0737, ITEM II.F.2 — Instrumentation for Detection of Inadequate Core Cooling (ICC) — Proposed License Condition 3 (TAC Numbers M77132 and M77133)

95-004

Disposition of NRC Technical Specification Open Item

TAC M76742-17

Open item - Page B3.5-5, LCO 3.5.1 - "2000 psig" in the second paragraph should be "1000 psig" to conform to the LCO.

Disposition

WBN agrees with the above item. The revised Technical Specification pages are included in Enclosure 4.

Disposition of NRC Technical Specification Open Item

TAC M76742-18

Open item - Page B 3.5-8, SR 3.5.1.5 - "2000 psig" in the first paragraph, third line, should be "1000 psig" to conform to the LCO.

Disposition

WBN agrees with the above item. The revised Technical Specification pages are included in Enclosure 4.

Disposition of NRC Technical Specification Open Item

TAC M76742-19

Open item - Page B3.6-27, SR 3.6.3.8 - The applicability of "[Bypass leakage is considered part of L_a]" needs to be determined for Watts Bar.

Disposition

Bypass leakage is not considered part of L_a. The attached markups indicate the changes necessary to address this item. The revised Technical Specification pages are included in Enclosure 4.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.8 (continued)

The frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions and therefore, the frequency extensions of SR 3.0.2 may not be applied since the testing is an Appendix J Type C test. This SR simply imposes additional acceptance criteria. *[Bypass leakage is considered part of L_a.]*

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis."
2. Watts Bar FSAR, Section 6.2.4.2, "Containment Isolation System Design," and Table 6.2.4-1, "Containment Penetrations and Barriers."
3. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."

THIS IS THE INFORMATION BEING CORRECTED

See Insert A

95-010

95-010 Insert A

Although not a part of L_a, the Shield Building Bypass leakage path combined leakage rate is determined using the 10CFR50 Appendix J Type B and C leakage rates for the applicable barriers.

Disposition of NRC Technical Specification Open Item

TAC M76742-20

Open item - Page B 3.7-13, LCO 3.7.3, Background - TVA needs to justify the deletion of the STS wording "and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops," from the end of the second paragraph.

Disposition

Watts Bar Unit 1 steam generators are preheat units, which have separate nozzles for the auxiliary feedwater supply. Therefore, MFIV isolation is not necessary to provide a pressure boundary for the controlled addition of auxiliary feedwater.

Disposition of NRC Technical Specification Open Item

TAC M76742-21

Open item - Page B 3.7-13, LCO 3.7.3, Background - TVA needs to justify the deletion of the following two sentences from the end of the third paragraph, "The AFW injection point is located downstream of the bypass MFIV so that AFW may be supplied to the steam generators following a bypass MFIV or MFRV closure. The piping volume between the check valve and the steam generators must be accounted for in calculating mass and energy release, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB."

Disposition

Watts Bar Unit 1 steam generators are preheat units, which have separate nozzles for the auxiliary feedwater supply. The deleted sentences are not applicable to the Watts Bar preheat steam generator design and therefore, should remain deleted.

Disposition of NRC Technical Specification Open Item

TAC M76742-22

Open item - Pages B 3.7-54 and B 3.7-55, LCO 3.7.10, Action D.1 - The proposed bases addition and changes to the succeeding Actions are dependent on resolution of Item 6 above.

Disposition

This item is addressed under TAC Item TAC M76742-6.

Disposition of NRC Technical Specification Open Item

TAC M76742-23

Open item - Page B 3.7-61, SR 3.7.11.1 - The wording in the fifth and sixth lines "main control room temperature is between 60°F and 104°F is not justified by the rest of the paragraph. The words should be replaced by the STS words "system has not degraded".

Disposition

The attached markup of the Westinghouse Technical Specification certification pages and the revised Technical Specification Bases pages address this item.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the sizing calculations in the control room. This SR consists of a combination of testing and calculations. This is accomplished by verifying that the system has not degraded. The only measurable parameters that could degrade undetected during normal operation is the system air flow and chilled water flow rate. Verification of these two flow rates will provide assurance that the heat removal capacity of the system is still adequate. The 18 month Frequency is appropriate since significant degradation of the CREATCS is slow and is not expected over this time period.

REFERENCES

1. Watts Bar FSAR, Section 9.4.1, "Control Room Area Ventilation System."

Disposition of NRC Technical Specification Open Item

TAC M76742-24

Open item - TR Pages 3.0-1 and 3.0-2, TR 3.0.4, SR 3.0.4; and various other pages and TR(s) - TVA needs to make similar changes and provide the necessary information to implement the changes made in Item 1 above.

Disposition

This item is addressed under TAC Item TAC M76742-1.

Disposition of NRC Technical Specification Open Item

TAC M76742-25

Open item - TR page 3.7-27, Table 3.7.5-1, Functions 29 through 32 - TVA needs to determine if an upper area temperature limit is applicable for the DG areas at Watts Bar.

Disposition

Upper temperature limits for the DG areas are provided on TR page 3.7-27, Table 3.7.5-1, Functions 21 and 22.

Disposition of NRC Technical Specification Open Item

TAC M76742-26

Open item - TR Pages B 3.0-5 through B 3.0-7, TR 3.0.4; Pages B 3.0-14 and B 3.0-15, TSR 3.0.4; and various other pages and TR(s) - TVA needs to make the appropriate changes to implement the change made in Item 24 above.

Disposition

This item is addressed under TAC Item TAC M76742-1.

ENCLOSURE 2

UPDATED ROAD MAP

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

The following summary table lists the Watts Bar Technical Specifications (1985 draft) and the corresponding new Technical Specifications (11/94 draft) based on the Westinghouse Owners Group MERITS program. Specifications which have been relocated out of the Technical Specifications are identified with the destination for the relocated Technical Specification.

The table lists each of the 1985 Watts Bar specifications, in numerical order, and the disposition of the specification in the MERITS program. The relocated specifications will go to the Technical Requirements Manual and various plant controlled documents or programs, some of which are defined in the Administrative Controls section of the MERITS Technical Specifications. Some operating and surveillance requirements are relocated to plant procedures and surveillance procedures which can be updated to reflect changes in plant design and operating conditions through the 50.54 or 50.59 review process.

This Table is intended to be used as a tool to assist procedure writers to make the conversion for the old TS to the new TS and to assist the NRC in completion of an SER for the use of the Improved Technical Specifications for licensing Watts Bar Unit 1. It is not intended to identify all differences between the two Technical Specifications.

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
2.1 <u>SAFETY LIMITS</u>	2.1 <u>SAFETY LIMITS</u>
2.1.1 Reactor Core 2.1.2 RCS Pressure 2.1.1 & 2.1.2 Action	2.1 Safety Limits and 2.2 Safety Limit Violation retain all requirements. Safety Limit 2.1.1 & 2.1.2 violation reporting requirements have been moved from specification 6.7.1 to SL 2.2.
2.2 <u>LIMITING SAFETY SYSTEM SETTINGS</u>	
2.2.1 Reactor Trip System Instrumentation Setpoints	3.3.1 Reactor Trip System Instrumentation The entire specification 2.2.1 has been moved to 3.3.1 RTS Instrumentation.

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.1 <u>REACTIVITY CONTROL SYSTEMS</u>	3.1 <u>REACTIVITY CONTROL SYSTEMS</u>
3.1.1.1 Shutdown Margin - T_{avg} - >200°F.	3.1.1 Shutdown Margin (SDM) - T_{avg} >200°F.
SR 4.1.1.1.1.a	Surveillance Requirement 4.1.1.1.1a has been included in MERITS LCO 3.1.5, Rod Group Alignment Limits, Actions
SR 4.1.1.1.1.b	SR 3.1.7.2
SR 4.1.1.1.1.c	SR 3.1.7.1
SR 4.1.1.1.1.d	Confirmed during the performance of SR 3.1.3.1
SR 4.1.1.1.1.e	SR 3.1.1.1 but "Factors" are moved to the BASES.
SR 4.1.1.1.2	SR 3.1.3.1.

3.1.1.2 Shutdown Margin - T_{avg} - ≥200°F. SR 4.1.1.2.a SR 4.1.1.2.b	3.1.2 Shutdown Margin - T_{avg} - ≥200°F. Relocated to LCO 3.1.5 SR 3.1.2.1, but "Factors" are moved to the BASES.

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.1.1.3 Moderate Temperature - Coefficient SR 4.1.1.3.a SR 4.1.1.3.b	3.1.4 Moderate Temperature Coefficient Moderator temperature coefficient limits have been relocated to the COLR. SR 3.1.4.1 SR 3.1.4.2 & 3.1.4.3
3.1.1.4 Minimum Temperature for Criticality SR 4.1.1.4	3.4.2 RCS Minimum Temperature for Criticality SR 3.4.2.1
3.1.2.1 Flow Paths - Shutdown	Relocated to the TRM, TR 3.1.1
3.1.2.2 Flow Paths - Operating	Relocated to the TRM, TR 3.1.2
3.1.2.3 Charging Pumps - Shutdown	Relocated to the TRM, TR 3.1.3 Old SR 4.1.2.3.2 is addressed by LCO 3.4.12 and was not retained in the TRM. SR 4.1.2.3.1 was revised.
3.1.2.4 Charging Pumps - Operating	Relocated to the TRM, TR 3.1.4 SR revised consistent with ECCS SR.
3.1.2.5 Borated Water Source Shutdown	Relocated to the TRM, TR 3.1.5
3.1.2.6 Borated Water Source Operating	Relocated to the TRM, TR 3.1.6

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.1.3.1 Moveable Control Assemblies - Group Height SR 4.1.3.1.1 SR 4.1.3.1.2 Table 3.1-1	3.1.5 Rod Group Alignment Limits SR 3.1.5.1 SR 3.1.5.2 Relocated to Bases text.
3.1.3.2 Position Indicating System - Operating SR 4.1.3.2	3.1.8 Rod Position Indication SR 3.1.5.1 SR 3.1.8.1 (added)
3.1.3.3 Position Indication System - Shutdown	Relocated to the TRM, TR 3.1.7
3.1.3.4 Rod Drop Time SR 4.1.3.4.a SR 4.1.3.4.b SR 4.1.3.4.c	Requirements of LCO are now covered by an SR in LCO 3.1.5. SR 3.1.5.3 Requirements covered by post-maintenance testing requirements SR 3.1.4.3
3.1.3.5 Shutdown Rod Insertion Limits SR 4.1.3.5	3.1.6 Shutdown Bank Insertion Limits SR 3.1.6.1
3.1.3.6 Control Rod Insertion Limits Figure 3.1-1 SR 4.1.3.6	3.1.7 Control Bank Insertion Limits The actual insertion limits have been relocated to the COLR. SR 3.1.7.2 SR 3.1.7.3 added

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.2 <u>POWER DISTRIBUTION LIMITS</u>	3.2 <u>POWER DISTRIBUTION LIMITS</u>
3.2.1 Axial Flux Difference (CAOC Methodology)	3.2.3 Axial Flux Difference (AFD) (RAOC Methodology)
Figure 3.2-1	The Watts Bar specific AFD target band has been relocated to the COLR. A <u>generic</u> AFD target band can be found in Figure B 3.2.3-1.
SR 4.2.1.1 SR 4.2.1.2 SR 4.2.1.3 SR 4.2.1.4	SR 3.2.3.1 Only one Surveillance exists due to the change in methodology (CAOC to RAOC).
3.2.2 Heat Flux Hot Channel Factor - $F_Q(Z)$	3.2.1 Heat Flux Hot Channel Factor - $F_Q(Z)$
	Allowable $F_Q(Z)$ vs. Power has been relocated to the COLR.
SR 4.2.2.1	Replaced with NOTE defining plant conditions for performing Srs.
SR 4.2.2.2	SR 3.2.1.1 & 3.2.1.2 Detailed $F^w{}_Q(Z)$ and $F^{Cq}(Z)$ measurement limitations have been moved to the BASES.
SR 4.2.2.3	Measurement uncertainty requirements have been moved to the BASES.
Figure 3.2-2	Relocated to the BASES.

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.2.3 RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor	3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)
	3.4.1 RCS Pressure, Temperature, and Flow DNB Limits The $F_{\Delta H}$ limits have been relocated to the COLR. Total flow vs. R has been deleted to reflect the latest methodology which reduces the available DNBR margin to offset rod bow penalty.
SR 4.2.3.1 SR 4.2.3.2 SR 4.2.3.3 SR 4.2.3.4 SR 4.2.3.5	SR 4.2.3.1 is included as a NOTE SR 3.2.2.1 SR 3.4.1.3 SR is captured by LCO 3.3.1 (RTS) SR 3.4.1.4
3.2.4 Quadrant Power Tilt Ratio SR 4.2.4.1 SR 4.2.4.2	3.2.4 Quadrant Power Tilt Ratio (QPTR) SR 3.2.4.1 SR 3.2.4.2
3.2.5 DNB Parameters SR 4.2.5	3.4.1 RCS Pressure, Temperature, and Flow DNB Limits SR 3.4.1.1 and 3.4.1.2

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.3 <u>INSTRUMENTATION</u>	3.3 <u>INSTRUMENTATION</u>
3.3.1 Reactor Trip System Instrumentation	3.3.1 Reactor Trip System (RTS) Instrumentation
Table 3.3-2 Reactor Trip System Instrumentation Response Times	Relocated to the TRM, TR 3.3.1
SR 4.3.1.1 SR 4.3.1.2	SR 3.3.1.1 through 3.3.1.14 SR 3.3.1.15
3.3.2 Engineered Safety Features Actuation System Instrumentation	3.3.2 Engineered Safety Features Actuation System (ESFAS) Instrumentation
Table 3.3-3, Item 3C, Containment Ventilation Isolation	Relocated to LCO 3.3.6 Containment Vent Isolation Instrumentation
Table 3.3-3, Item 8, 6.9kV Shutdown Board	Relocated to LCO 3.3.5
Table 3.3-5 Engineered Safety Feature Response Times	Relocated to the TRM, TR 3.3.2
SR 4.3.2.1	SR 3.3.2.1 through 3.3.2.9 and 3.3.2.11
SR 4.3.2.2	SR 3.3.2.10

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.3.3.1 Radiation Monitoring for Plant Operations SR 4.3.3.1	3.4.15 RCS Leakage Detection Instrumentation 3.3.7 Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation 3.3.8 Auxiliary Building Gas Treatment System (ABGTS) Actuation Instrumentation Containment Atmosphere Radiation Monitors replaced by Containment Purge Exhaust Radiation Monitors SR 3.4.15.1, 3.4.15.2, and 3.4.15.4 SR 3.3.7.1, 3.3.7.2, and 3.3.7.4 SR 3.3.8.1, 3.3.8.2, and 3.3.8.4 Added SR 3.4.15.3, 3.3.7.3, and 3.3.8.3 to perform TADOT.
3.3.3.2 Movable Incore Detectors	Relocated to the TRM, TR 3.3.3
3.3.3.3 Seismic Instrumentation	Relocated to the TRM, TR 3.3.4
3.3.3.4 Meteorological Instrumentation	Relocated to the Offsite Dose Calculation Manual (ODCM)
3.3.3.5 Remote Shutdown Instrumentation SR 4.3.3.5 Table 3.3-9	3.3.4 Remote Shutdown System SR 3.3.4.1 through 3.3.4.4 Relocated to LCO 3.3.4 Table 3.3.4-1
3.3.3.6 Accident Monitoring Instrumentation SR 4.3.3.6	3.3.3 Post-Accident Monitoring (PAM) Instrumentation SR 3.3.3.1 thru 3.3.3.4
3.3.3.7 Fire Detection Instrumentation	Relocated to the Fire Protection Plan (FPP)
3.3.3.8 Radioactive Liquid Effluent Monitoring Instrumentation	Relocated to the ODCM

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.3.3.9 Radioactive Gaseous Effluent Monitoring Instrumentation	Relocated to the ODCM
3.3.3.10 Loose-Part Detection System	Relocated to the TRM, TR 3.3.6
3.3.4 Turbine Overspeed Protection	Relocated to the TRM, TR 3.3.5
3.4 <u>REACTOR COOLANT SYSTEM</u>	3.4 <u>REACTOR COOLANT SYSTEM (RCS)</u>
3.4.1.1 RCS Startup and Power Operation SR 4.4.1.1	3.4.4 RCS Loops - MODES 1 and 2 SR 3.4.4.1
3.4.1.2 RCS Hot Standby SR 4.4.1.2.1 SR 4.4.1.2.2 SR 4.4.1.2.3	3.4.5 RCS Loops - MODE 3 SR 3.4.5.3 SR 3.4.5.2 SR 3.4.5.1
3.4.1.3 RCS Hot Shutdown SR 4.4.1.3.1 SR 4.4.1.3.2 SR 4.4.1.3.3	3.4.6 RCS Loops - MODE 4 SR 3.4.6.3 SR 3.4.6.2 SR 3.4.6.1
3.4.1.4.1 RCS Cold Shutdown - Loops Filled SR 4.4.1.4.1.1 SR 4.4.1.4.1.2	3.4.7 RCS Loops - MODE 5, Loops Filled SR 3.4.7.2 SR 3.4.7.1 SR 3.4.7.3 added for RHR pump breaker alignment and power
3.4.1.4.2 RCS Cold Shutdown - Loops Not Filled SR 4.4.1.4.2	3.4.8 RCS Loops - MODE 5, Loops Not Filled SR 3.4.8.1 SR 3.4.8.2 added for RHR pump breaker alignment and power for the non-operating pump.
3.4.2.1 Safety Valves - Shutdown	Relocated to TRM, TR 3.4.1

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.4.2.2 Safety Valves - Operating SR 4.4.2.2	3.4.10 Pressurizer Safety Valves SR 3.4.10.1
3.4.3 Pressurizer SR 4.4.3.1 SR 4.4.3.2 SR 4.4.3.3	3.4.9 Pressurizer SR 3.4.9.1 SR 3.4.9.2 Deleted, not required since always powered by 1E power
3.4.4 Relief Valves SR 4.4.4.1.a SR 4.4.4.1.b SR 4.4.4.2 SR 4.4.4.3	3.4.11 Pressurizer Power-Operated Relief Valves (PORVs) SR 3.4.12.8 SR 3.4.11.2 SR 3.4.11.1 Deleted, not required since always powered by 1E power
3.4.5 Steam Generators SR 4.4.5.0 through 4.4.5.5	Requirements covered by SR 3.4.4.2. Relocated to Steam Generator Tube, Surveillance Program Specification 5.7.2.12.
3.4.6.1 Leakage Detection Systems SR 4.4.6.1	3.4.15 RCS Leakage Detection Instrumentation SR 3.4.15.1 through 3.4.15.4

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)		WATTS BAR MERITS, Title or Disposition
3.4.6.2	Operational Leakage	3.4.13 RCS Operational Leakage 3.5.5 Seal Injection Flow 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage
	SR 4.4.6.2.1.a	SR 3.4.15.1
	SR 4.4.6.2.1.b	SR 3.4.15.3
	SR 4.4.6.2.1.c	Relocated to LCO 3.5.5
	SR 4.4.6.2.1.d	SR 3.4.13.1
	SR 4.4.6.2.1.e	Deleted
	SR 4.4.6.2.2.a SR 4.4.6.2.2.b	SR 3.4.14.1
	SR 4.4.6.2.2.c	Requirements covered by post-maintenance test requirements.
	SR 4.4.6.2.2.d	SR 3.4.14.1
3.4.7	Chemistry	Relocated to TRM, TR 3.4.4
3.4.8	Specific Activity	3.4.16 RCS Specific Activity
	SR 4.4.8	SR 3.4.16.1 through 3.4.16.3
3.4.9.1	Pressure/Temperature Limits	3.4.3 RCS Pressure and Temperature (P/T) Limits
	RCS heatup and cooldown curves (Figures 3.4-2 and 3.4-3)	Relocated to RCS Pressure and Temperature Limits Report (PTLR) per specification 5.9.6
	Reactor Vessel Material Surveillance Program - Withdrawal Schedule (Table 4.4-5)	Relocated to PTLR
	SR 4.4.9.1.1	SR 3.4.3.1
	SR 4.4.9.1.2	Relocated to PTLR
3.4.9.2	Pressurizer Pressure/Temperature Limits	Relocated to the TRM, TR 3.4.2

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)		WATTS BAR MERITS, Title or Disposition
3.4.9.3	Overpressure Protection Systems	3.4.12 Cold Overpressure Mitigation System (COMS) PORV lift settings relocated to PTLR
	SR 4.4.9.3.1.a	SR 3.4.12.7
	SR 4.4.9.3.1.b	SR 3.4.12.8
	SR 4.4.9.3.1.c	SR 3.4.12.5
	SR 4.4.9.3.2	SR 3.4.12.4
3.4.10	Structural Integrity	Relocated to TRM, TR 3.4.5
3.4.11	Reactor Coolant System Vents	Relocated to the TRM, TR 3.4.3
3.5	<u>EMERGENCY CORE COOLING SYSTEMS</u>	3.5 <u>EMERGENCY CORE COOLING SYSTEMS (ECCS)</u>
3.5.1.1	Accumulators Cold Leg Injection	3.5.1 Accumulators
	SR 4.5.1.1.1	SR 3.5.1.1 through 3.5.1.5
	SR 4.5.1.1.2	No longer required by new STS NUREG-1431
3.5.1.2	Upper Head Injection	UHI System deleted

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)		WATTS BAR MERITS, Title or Disposition
3.5.2	ECCS Subsystems - T_{avg} $\geq$ 350°F	3.5.2 ECCS - Operating
	Action B	See C.20
	SR 4.5.2.a	SR 3.5.2.1
	SR 4.5.2.b	SR 3.5.2.2 and 3.5.2.3
	SR 4.5.2.c	PAI-2.03
	SR 4.5.2.d.1.a	No longer required due to elimination of Auto Closure Interlock (ACI) from design.
	SR 4.5.2.d.1.b	No longer required due to elimination of Auto Closure Interlock (ACI) from design.
	SR 4.5.2.d.2	SR 3.5.2.8
	SR 4.5.2.e	SR 3.5.2.5 and 3.5.2.6
	SR 4.5.2.f	SR 3.5.2.4, actual pump heads are specified per Inservice Testing Program per specification 5.7.2.11
	SR 4.5.2.g	SR 3.5.2.7, the 4-hour requirement was relocated since the requirement was actually a maintenance activity
	SR 4.5.2.h	Requirements covered by post-maintenance requirements
3.5.3	ECCS Subsystems - T_{avg} $\geq$ 350°F	3.5.3 ECCS - Shutdown
	Action C	See C.20
	SR 4.5.3.1	SR 3.5.3.1
	SR 4.5.3.2	SR 3.4.12.1 & SR 3.4.12.2
3.5.4	Refueling Water Storage Tank	3.5.4 Refueling Water Storage Tank (RWST)
	SR 4.5.4.a	SR 3.5.4.2 and 3.5.4.3
	SR 4.5.4.b	SR 3.5.4.1

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)		WATTS BAR MERITS, Title or Disposition	
3.6	CONTAINMENT SYSTEMS	3.6	CONTAINMENT SYSTEMS
3.6.1.1	Containment Integrity SR 4.6.1.1.a SR 4.6.1.1.b SR 4.6.1.1.c	3.6.1	Containment SR 3.6.3.2 and 3.6.3.3 LCO 3.6.2 SR 3.6.1.1 and 10CFR 50, Appendix J requirements
3.6.1.2	Containment Leakage		LCO requirements addressed by LCO 3.6.1 and LCO 3.6.3, SR 3.6.1.1, SR 3.6.3.8 and 10CFR50 Appendix J
3.6.1.3	Containment Air Locks SR 4.6.1.3.a SR 4.6.1.3.b SR 4.6.1.3.c	3.6.2	Containment Air Locks SR 3.6.2.1 SR 3.6.2.2 and 10 CFR50, Appendix J
3.6.1.4	Internal Pressure SR 4.6.1.4	3.6.4	Containment Pressure SR 3.6.4.1
3.6.1.5	Air Temperature SR 4.6.1.5.1 SR 4.6.1.5.2	3.6.5	Containment Air Temperature SR 3.6.5.1 SR 3.6.5.2
3.6.1.6	Containment Vessel Structural Integrity		Addressed by SR 3.6.1.1
3.6.1.7	Shield Building Structural Integrity SR 4.6.1.7	3.6.15	Shield Building SR 3.6.15.3

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.6.1.8 Emergency Gas Treatment System	3.6.9 Emergency Gas Treatment System (EGTS)
SR 4.6.1.8.1.a SR 4.6.1.8.1.b SR 4.6.1.8.1.c SR 4.6.1.8.1.d.1 SR 4.6.1.8.1.d.5 SR 4.6.1.8.1.e SR 4.6.1.8.1.f	SR 3.6.9.1 SR 3.6.9.2 and Ventilation Filter Testing Program per specification 5.7.2.14
SR 4.6.1.8.1.d.2	SR 3.6.9.3
SR 4.6.1.8.1.d.3	Open - May be relocated to plant surveillance
SR 4.6.1.8.1.d.4	SR 3.6.15.1 & SR 3.6.15.4
SR 4.6.1.8.2	SR 3.6.15.1
3.6.1.9 Containment Ventilation System	Requirements addressed as shown below
SR 4.6.1.9.1	SR 3.6.3.1 & SR 3.6.3.7
SR 4.6.1.9.2	No longer required by new STS NUREG-1431
SR 4.6.1.9.3	SR 3.6.3.5 changed frequency to 184 days and within 92 days after opening valve
3.6.2 Containment Spray System	3.6.6 Containment Spray System (CSS)
SR 4.6.2.a	SR 3.6.6.1
SR 4.6.2.b	SR 3.6.6.2 and Inservice Testing Program per specification 5.7.2.11
SR 4.6.2.c	SR 3.6.6.3 and 3.6.6.4
SR 4.6.2.d	SR 3.6.6.5
	Added SR 3.6.6.6 to perform SR 3.5.2.2 and SR 3.5.2.4 for the RHR spray system.
3.6.3 Containment Isolation Valves	3.6.3 Containment Isolation Valves
SR 4.6.3.1	Requirements covered by post-maintenance requirements
SR 4.6.3.2	SR 3.6.3.6
SR 4.6.3.3	SR 3.6.3.4
Table 3.6-2	Covered by FSAR 6.2.4-1

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.6.4.1 Hydrogen Monitors	Relocated to LCO 3.3.3
3.6.4.2 Electric Hydrogen Recombines SR 4.6.4.2.a SR 4.6.4.2.b.1 SR 4.6.4.2.b.2 SR 4.6.4.2.b.3	3.6.7 Hydrogen Recombiners - MODES 1 and 2 SR 3.6.7.1 No longer required by new STS NUREG-1431 SR 3.6.7.2 SR 3.6.7.3
3.6.4.3 Hydrogen Mitigation System SR 4.6.4.3	3.6.8 Hydrogen Ignition System (HIS) SR 3.6.8.1 and 3.6.8.3 Added SR 3.6.8.2 to verify ≥ one ignitor OPERABLE in each region
3.6.5.1 Ice Bed SR 4.6.5.1	3.6.11 Ice Bed SR 3.6.11.1 through 3.6.11.6
3.6.5.2 Ice Bed Temperature Monitoring System	Relocated to the TRM, TR 3.6.1
3.6.5.3 Ice Condenser Doors SR 4.6.5.3.1.a SR 4.6.5.3.1.b.1 SR 4.6.5.3.1.b.2 SR 4.6.5.3.1.b.3 SR 4.6.5.3.1.b.4 SR 4.6.5.3.1.b.5 SR 4.6.5.3.2 SR 4.6.5.3.3	3.6.12 Ice Condenser Doors SR 3.6.12.1 SR 3.6.12.4 SR 3.6.12.3 SR 3.6.12.5, detailed acceptance criteria relocated to the Bases. SR 3.6.12.2 and 3.6.12.6, detailed acceptance criteria relocated to the Bases. SR 3.6.12.7
3.6.5.4 Inlet Door Position Monitoring System	Relocated to the TRM, TR 3.6.2
	New TR 3.6.3 Lower Compartment Cooling System added to be consistent with SNP.

WATTS BAR Draft, Title (LCO and Surveillance)		WATTS BAR MERITS, Title or Disposition	
3.6.5.5	Divider Barrier, Personnel Access Doors and Equipment Hatches SR 4.6.5.5.1 SR 4.6.5.5.2	3.6.13	Divider Barrier Integrity SR 3.6.13.1 and 3.6.13.3 SR 3.6.13.2
3.6.5.6	Containment Air Return Fan Systems SR 4.6.5.6	3.6.10	Air Return System (ARS) SR 3.6.10.1 through 3.6.10.3
3.6.5.7	Floor Drains SR 4.6.5.8	3.6.14	Containment Recirculation Drains SR 3.6.14.2
3.6.5.8	Refueling Canal Drains SR 4.6.5.8	3.6.14	Containment Recirculation Drains SR 3.6.14.1
3.6.5.9	Divider Barrier Seal SR 4.6.5.9 Table 3.6-3	3.6.13	Divider Barrier Integrity SR 3.6.13.4 and 3.6.13.5 Included in SR 3.6.13.4
3.7	<u>PLANT SYSTEMS</u>	3.7	<u>PLANT SYSTEMS</u>
3.7.1	Turbine Cycle Safety Valves SR 4.7.1.1	3.7.1	Main Steam Safety Valves (MSSVs) SR 3.7.1.1
3.7.1.2	Auxiliary Feedwater System SR 4.7.1.2.1.a.1 SR 4.7.1.2.1.a.2 SR 4.7.1.2.1.a.3 SR 4.7.1.2.1.a.4 SR 4.7.1.2.1.b.1 SR 4.7.1.2.1.b.2 SR 4.7.1.2.2	3.7.5	Auxiliary Feedwater (AFW) System SR 3.7.5.2 and Inservice Testing Program, per specification 5.7.2.11 SR 3.7.5.1 SR 3.7.5.5 SR 3.7.5.3 SR 3.7.5.4 SR 3.7.5.5
3.7.1.3	Condensate Storage Tank SR 4.7.1.3.1 SR 4.7.1.3.2	3.7.6	Condensate Storage Tank (CST) SR 3.7.6.1 LCO 3.7.8

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.7.1.4 Specific Activity SR 4.7.1.4 Table 4.7-1	3.7.14 Secondary Specific Activity SR 3.7.14.1
3.7.1.5 Main Steam Line Isolation Valves SR 4.7.1.5	3.7.2 Main Steam Line Isolation Valves (MSIVs) SR 3.7.2.1 and Inservice Testing Program per specification 5.7.2.11
3.7.2 Steam Generator Pressure/Temperature Limitation	Relocated to the TRM, TR 3.7.1
3.7.3 Component Cooling Water System SR 4.7.3.a SR 4.7.3.b SR 4.7.3.c	3.7.7 Component Cooling Water System (CCS) SR 3.7.7.2 SR 3.7.7.4 SR 3.7.7.1 SR 3.7.7.3 added to verify automatic valve actuation
3.7.4 Essential Raw Cooling Water System SR 4.7.4	3.7.8 Essential Raw Cooling Water System (ERCW) SR 3.7.8.1 through 3.7.8.3
3.7.5 Ultimate Heat Sink SR 4.7.5	3.7.9 Ultimate Heat Sink (UHS) SR 3.7.9.1
3.7.6 Flood Protection Plan	Relocated to the TRM, TR 3.7.2

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.7.7 Control Room Emergency Ventilation System	3.7.10 Control Room Emergency Ventilation System (CREVS) 3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)
SR 4.7.7.a	Covered by TRM 3.7.5, Area Temperature Monitoring
SR 4.7.7.b	SR 3.7.10.1 and Ventilation Filter Testing Program per specification 5.7.2.14
SR 4.7.7.c SR 4.7.7.d SR 4.7.7.e.1 SR 4.7.7.f SR 4.7.7.g	SR 3.7.10.2 and Ventilation Filter Testing Program per specification 5.7.2.14
SR 4.7.7.e.2 SR 4.7.7.e.3	SR 3.7.10.3 SR 3.7.10.4

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.7.8 Auxiliary Building Gas Treatment System SR 4.7.8.a SR 4.7.8.b SR 4.7.8.c SR 4.7.8.d.1 SR 4.7.8.d.4 SR 4.7.8.e SR 4.7.8.f SR 4.7.8.d.2 SR 4.7.8.d.3	3.7.12 Auxiliary Building Gas Treatment System (ABGTS) SR 3.7.12.1 SR 3.7.12.2 and Ventilation Filter Testing Program per specification 5.7.2.14 SR 3.7.12.3 SR 3.7.12.4
3.7.9 Snubbers	Relocated to the TRM, TR 3.7.3
3.7.10 Sealed Source	Relocated to the TRM, TR 3.7.4
3.7.11 Fire Suppression Water System	Relocated to the FPP
3.7.11.2 Spray and/or Sprinkler Systems	Relocated to the FPP
3.7.11.3 CO_2 Systems	Relocated to the FPP
3.7.11.4 Fire Hose Stations	Relocated to the FPP
3.7.12 Fire Related Assemblies	Relocated to the FPP
3.7.13 Area Temperature Monitoring	Relocated to the TRM, TR 3.7.5
N/A	3.7.3 Main Feedwater Isolation and Regulation Valves (MFIVs and MFRVs) and Associated Bypass Valves LCO added since Main Feed isolated is assumed in the safety analyses.
N/A	3.7.4 Atmospheric Dump Valves (ADVs) LCO added since ADVs are assumed in the safety analysis for SGTR
3.8 <u>ELECTRICAL POWER SYSTEMS</u>	3.8 <u>ELECTRICAL POWER SYSTEMS</u>

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.8.1.1 A.C. Sources Operating	3.8.1 AC Sources - Operating 3.8.3 Diesel Fuel and Lubricating Oil
SR 4.8.1.1.1.a	SR 3.8.1.1
SR 4.8.1.1.1.b	SR 3.8.1.8
SR 4.8.1.1.2.a.1	SR 3.8.1.4
SR 4.8.1.1.2.a.2	SR 3.8.3.1
SR 4.8.1.1.2.a.3	SR 3.8.1.6
SR 4.8.1.1.2.a.4	SR 3.8.1.2 with voltage/frequency values relaxed to steady-state limits, and SR 3.8.1.7 at reduced frequency requirements
SR 4.8.1.1.2.a.5	SR 3.8.1.3
SR 4.8.1.1.2.a.6	Covered by definition of operability
SR 4.8.1.1.2.b	SR 3.8.1.5
SR 4.8.1.1.2.c	SR 3.8.3.5
SR 4.8.1.1.2.d.1 SR 4.8.1.1.2.d.2 SR 4.8.1.1.2.e	SR 3.8.3.3 and the Diesel Fuel Oil Testing Program Specification 5.7.2.16
SR 4.8.1.1.2.f.1	Requirements covered by post-maintenance requirements
SR 4.8.1.1.2.f.2	SR 3.8.1.9
SR 4.8.1.1.2.f.3	SR 3.8.1.10
SR 4.8.1.1.2.f.4	SR 3.8.1.11
SR 4.8.1.1.2.f.5	SR 3.8.1.12
SR 4.8.1.1.2.f.6.a,b	SR 3.8.1.19
	SR 3.8.1.13
SR 4.8.1.1.2.f.6.c	SR 3.8.1.14 and 3.8.1.15
SR 4.8.1.1.2.f.7	

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.8.1.1 (continued)	
SR 4.8.1.1.2.f.8	Covered by 10 CFR 50.59
SR 4.8.1.1.2.f.9	SR 3.8.1.16
SR 4.8.1.1.2.f.10	SR 3.8.1.18
SR 4.8.1.1.2.f.11	Deleted
SR 4.8.1.1.2.f.12	SR 3.8.3.6
SR 4.8.1.1.2.g	SR 3.8.1.21
SR 4.8.1.1.2.h	SR 3.8.3.7
SR 4.8.1.1.3.a.1	SR 3.8.4.4
SR 4.8.1.1.3.a.2	SR 3.8.6.1
SR 4.8.1.1.3.a.3	SR 3.8.4.2
SR 4.8.1.1.3.b.1	SR 3.8.6.2
SR 4.8.1.1.3.b.2	SR 3.8.4.6
SR 4.8.1.1.3.b.3	SR 3.8.6.3
SR 4.8.1.1.3.c.1	SR 3.8.4.7
SR 4.8.1.1.3.c.2	SR 3.8.4.8
SR 4.8.1.1.3.c.3	SR 3.8.4.10
SR 4.8.1.1.3.d	SR 3.8.4.13
SR 4.8.1.1.3.e	SR 3.8.4.14
SR 4.8.1.1.3.d	SR 3.8.4.14
SR 4.8.1.1.4	Reports relocated to specification 5.9.7.
	SR 3.8.3.2, 3.8.3.4, and 3.8.3.6 added
3.8.1.2 A.C. Sources Shutdown	3.8.3 A.C. Sources - Shutdown
SR 4.8.1.2	SR 3.8.2.1
3.8.2.1 D.C. Sources Operating	3.8.4 DC Sources - Operating 3.8.6 Battery Cell Parameters
SR 4.8.2.1.a.1	SR 3.8.6.1
SR 4.8.2.1.a.2	SR 3.8.4.1
SR 4.8.2.1.a.3	SR 3.8.4.3
SR 4.8.2.1.b.1	SR 3.8.6.2
SR 4.8.2.1.b.2	SR 3.8.4.5
SR 4.8.2.1.b.3	SR 3.8.6.3
SR 4.8.2.1.c.1	SR 3.8.4.7
SR 4.8.2.1.c.2	SR 3.8.4.8
SR 4.8.2.1.c.3	SR 3.8.4.9
SR 4.8.2.1.c.4	SR 3.8.4.11
SR 4.8.2.1.d	SR 3.8.4.13
SR 4.8.2.1.e	SR 3.8.4.14
SR 4.8.2.1.f	SR 3.8.4.14

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.8.2.2 D.C. Sources Shutdown SR 4.8.2.2	3.8.5 DC Sources - Shutdown SR 3.8.5.1
3.8.3.1 Onsite Power Distribution Operating SR 4.8.3.1.1 SR 4.8.3.1.2	3.8.9 Distribution Systems - Operating 3.8.7 Inverters - Operating SR 3.8.9.1, 3.8.7.1 Requirements covered by TR 3.7.5, Area Temperature Monitoring
3.8.3.2 Onsite Power Distribution Shutdown SR 4.8.3.2	3.8.10 Distribution Systems - Shutdown 3.8.8 Inverters - Shutdown SR 3.8.10.1, 3.8.8.1
3.8.3.3 Isolation Devices	Relocated to the TRM, TR 3.8.1
3.8.4.1 Containment Penetration Conductor Overcurrent Protective Devices	Relocated to the TRM, TR 3.8.2
3.8.4.2 Motor-Operated Valves Thermal Overload Bypass Devices	Relocated to the TRM, TR 3.8.3
3.8.4.3 Submerged Component Circuit Protection	Relocated to the TRM, TR 3.8.4
3.9 <u>REFUELING OPERATIONS</u>	3.9 <u>REFUELING OPERATIONS</u>
3.9.1 Boron Concentration SR 4.9.1.1 SR 4.9.1.2 SR 4.9.1.3	3.9.1 Boron Concentration 3.9.2 Unborated Water Source Isolated Valves SR 3.9.1.1 The required boron concentration and sampling conditions relocated to the COLR. The K_{eff} requirement has been deleted because the boron concentration requirement is more restrictive and, if met, guarantees that the K_{eff} requirement is met. SR 3.9.2.1 valve list relocated to surveillance program (I-SI-62-1)

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.9.2 Instrumentation SR 4.9.2.a SR 4.9.2.b SR 4.9.2.c	3.9.3 Nuclear Instrumentation SR 3.9.3.1 SR 3.3.1.8 SR 3.9.3.2
3.9.3 Decay Time	Relocated to TRM, TR 3.9.1
3.9.4 Containment Building Penetrations SR 4.9.4.a SR 4.9.4.b	3.9.4 Containment Building Penetrations SR 3.9.4.1 SR 3.9.4.2
3.9.5 Communications	Relocated to TRM, TR 3.9.2
3.9.6 Refueling Machine	Relocated to TRM, TR 3.9.3
3.9.7 Crane Travel - Spent Fuel Storage Pool Building	Relocated to TRM, TR 3.9.4
3.9.8.1 RHR and Coolant Circulation High Water Level SR 4.9.8.1	3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level SR 3.9.5.1
3.9.8.2 RHR and Coolant Circulation Low Water Level SR 4.9.8.2	3.9.6 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level SR 3.9.6.1
3.9.9 Containment Ventilation Isolation System	Relocated to LCO 3.9.4 Containment Penetrations
3.9.10 Water Level - Reactor Vessel SR 4.9.10	3.9.7 Refueling Cavity Water Level SR 3.9.7.1
3.9.11 Water Level - Storage Pool	3.7.13 Fuel Storage Pool Water Level Applicability changed to during movement of irradiated fuel assemblies

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
3.9.12 Auxiliary Building Gas Treatment System	3.7.12 Auxiliary Building Gas Treatment System Applicability changed to MODES 1, 2, 3, and 4 and during movement of irradiated fuel assemblies
3.9.13 Reactor Building Purge Ventilation System	3.9.8 Reactor Building Purge Air Cleanup Units This Surveillance has been moved to the Ventilation Filter Testing Program 5.7.2.14
3.10 <u>SPECIAL TEST EXCEPTIONS</u>	N/A
3.10.1 Shutdown Margin SR 4.10.1.1 SR 4.10.1.2	3.1.10 Physics Test Exceptions - MODE 2
3.10.2 Group Height, Insertion, and Power Distribution Limits SR 4.10.2.1 SR 4.10.2.2	3.1.9 Physics Test Exceptions - MODE 1 SR 3.1.9.1 SR 3.1.9.3 SR 3.1.9.2 added to verify Power Range Neutron Flux-High trip setpoints Added SR 3.1.9.4 to verify SDM $\geq$ 1.6 $\Delta k/k$
3.10.3 Physics Tests SR 4.10.3.1 SR 4.10.3.2 SR 4.10.3.3	3.1.10 Physics Test Exceptions - MODE 2 Covered by MODE definition SR 3.1.10.1 SR 3.1.10.2 Added SR 3.1.10.3 to verify SDM $\geq$ 1.6 $\Delta k/k$

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)		WATTS BAR MERITS, Title or Disposition	
3.10.4	Reactor Coolant Loops SR 4.10.4.1 SR 4.10.4.2	3.4.17	RCS Loops - Test Exceptions SR 3.4.17.1 SR 3.4.17.2
3.11	RADIOACTIVE EFFLUENT		
3.11.1.1	Liquid Effluent	Relocated to the ODCM	
3.11.1.2	Dose	Relocated to the ODCM	
3.11.1.3	Liquid Radwaste Treatment System	Relocated to the ODCM	
3.11.1.4	Liquid Holdup Tanks	Relocated to the ODCM	
3.11.2.1	Gaseous Effluent	Relocated to the ODCM	
3.11.2.2	Dose - Noble Gases	Relocated to the ODCM	
3.11.2.3	Dose - Iodine-131 and 133, Tritium, and Radioactive Material in Particulate Form	Relocated to the ODCM	
3.11.2.4	Gaseous Radwaste Treatment System	Relocated to the ODCM	
3.11.2.5	Explosive Gas Mixture	Relocated to the Explosive Gas and Storage Tank Radioactivity Monitoring Program per specification 5.7.2.15	
3.11.2.6	Gas Decay Tanks	Relocated to the ODCM	
3.11.3	Solid Radioactive Wastes	Relocated to the Process Control Program (PCP)	
3.11.4	Total Dose	Relocated to the ODCM	
3.12	RADIOLOGICAL ENVIRONMENTAL MONITORING		
3.12.1	Monitoring Program	Relocated to the ODCM	
3.12.2	Land Use Census	Relocated to the ODCM	
3.12.3	Interlaboratory Comparison Program	Relocated to the ODCM	

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)		WATTS BAR MERITS, Title or Disposition	
5.0	Design Features	4.0	Design Features
5.1	Site	4.1	Site
5.2	Containment	Addressed by LCO 3.6.1 Containment, LCO 3.6.4 Containment Pressure, LCO 3.6.5 Containment Air Temperature, and LCO 3.6.15 Shield Building	
5.3	Reactor Core	4.2	Reactor Core
5.4.1.a	Reactor Coolant System Design Pressure and Temperature	Addressed by 5.7.2.11 Inservice Testing Program, 5.7.2.10 Reactor Coolant Pump Flywheel Inspection Program, 5.9.6 RCS PTLR, and TR 3.4.5 Piping System Structural Integrity	
5.4.2	Reactor Coolant System Volume	Addressed by LCO 3.4.9 Pressurizer	
5.5	Meteorological Tower Location	FSAR Section 2.3.3.1	
5.6	Fuel Storage	4.3	Fuel Storage
5.7	Component Cyclic or Transient Limit	5.7.2.9	Component Cyclic or Transient Limit
6.1	Responsibility	5.1	Responsibility
6.1.1	Plant Manager	5.1.1 (See C.1)	
6.1.2	Manager Radiological Health	5.7.2.3	Offsite Dose Calculation Manual (ODCM) and TVA-NPOD89-A Section 4.2.5.A.4
6.1.3	Shift Supervisor	5.1.2 (See C.2)	
6.2	Organization	5.2	Organization
6.2.1	Offsite organization management and technical support	TVA-NPOD89-A	
6.2.2.a	Minimum shift crew composition	See C.3	
6.2.2.b	Licensed operators in the Control Room	See C.3	
6.2.2.c	Health Physics Technician	5.2.2.d	

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
6.2.2.d CORE ALTERATIONS	See C.4
6.2.2.e Fire Brigade	Fire Protection Program
6.2.2.f Working hours	5.2.2.e
Fig. 6.2-1 Offsite Organization for Facility Management and Technical Support	TVA-NPOD89-A
Fig. 6.2-2 Facility Organization	TVA-NPOD89-A
Table 6.1-1 Surveillance Requirements Performed by Radiological Health	Relocated to the ODCM, PCP, or other administrative control programs as appropriate
Table 6.2-1 Minimum Shift Crew Composition	See C.3
6.2.3 Independent Safety Engineering Group (ISEG)	FSAR 13.4.2 and SSP-4.08
6.2.4 Shift Technical Advisor (STA)	5.5.2.g
6.3 Unit Staff Qualifications	5.3.1
6.4 Training	See C.5
6.5.0 Manager, Office of Nuclear Power	TVA-NPOD89-A
6.5.1 Plant Operations Review Committee (PORC)	FSAR 13.4.1 and SSP-12.54
6.5.2 Nuclear Safety Staff (NSS)	FSAR 13.4.2 and SSP-4.08
6.5.3 Radiological Assessment Review Committee (RARC)	Now covered by PORC
6.6 Reportable Event Action	10 CFR 50.73 and SSP-4.05
Safety Limit Violation 6.7.1.a, 6.7.1.b, 6.7.1.c, and 6.7.1.d	2.2
6.8.1.a	5.7.1.1.a
6.8.1.b	5.7.1.1.b
6.8.1.c	See C.6
6.8.1.d	See C.6

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)	WATTS BAR MERITS, Title or Disposition
6.8.1.e	See C.11
6.8.1.f	5.7.1.1.c
6.8.1.g	5.7.1.1.d
6.8.2, 6.8.3.a, 6.8.3.b, 6.8.3.c	SSP-2.03, SSP-2.04, Master Site Procedure List (MSPL), and TVA-NQA-PLN89-A See C.9
6.8.4.a	See C.9 and 5.7.2.3
6.8.4.b	See C.9 and 5.7.2.3
6.8.4.c	See C.9 and 5.7.2.3
6.8.5.a Reactor Coolant Sources Outside Containment	5.7.2.4
6.8.5.b In-Plant Radiation Monitoring	See C.12
6.8.5.c Secondary Water Chemistry	5.7.2.13
6.8.5.d Post-accident Sampling	5.7.2.6
6.8.5.e Turbine Integrity Program with Turbine Overspeed Protection (TIPTOP)	Relocated to TR 3.3.5
6.9.1 Routine Reports	5.9
6.9.1.1, 6.9.1.2, and 6.9.1.3, Startup Reports	See C.18
6.9.1.4 and 6.9.1.5, Annual Reports	5.9.1
6.9.1.6 Annual radiological Environmental Operating Report	5.9.2
6.9.1.7 Semiannual Radioactive Effluent Release Report	5.9.3
6.9.1.8 Monthly Operating Reports	5.9.4
6.9.1.9 Radial Peaking Factor Limit Report	5.9.5
6.9.2 Special Reports	See C.20
6.10 Record Retention	See C.21

WATTS BAR - RETENTION AND RELOCATION SUMMARY TABLE

WATTS BAR Draft, Title (LCO and Surveillance)		WATTS BAR MERITS, Title or Disposition
6.11	Radiation Protection Program	See C.10
6.12	High Radiation Area	5.11
6.13	Process Control Program	See C.11
6.14	Offsite Dose Calculation Manual (ODCM)	5.7.2.3
6.15	Major Changes to Liquid, Gaseous, and Solid Radwaste Treatment Systems	5.7.2.3

RELOCATION NOTES

C.1 The new Technical Specifications added a requirement that the Plant Manager approve, prior to implementation, each proposed test, experiment, or modifications to systems or equipment that affect nuclear safety.

C.2 TS need not require an administrative letter be issued to station personnel on an annual basis describing responsibility of the Shift Operations Supervisor (SOS). The SOS responsibilities are adequately described in TVA-NPOD89-A Section 4.2.5.A.3. Repeating the organization responsibilities via an internal management directive only increases the administrative burden on the facility with no resulting benefit. Plant safety is not compromised by this proposed change.

The markup reflects the change from the word "valid" to "active" in accordance with the recommendations of the NRC letter dated October 25, 1993.

C.3 The minimum Shift Crew Composition Table need not be retained in TS. 10CFR 50.54(k), (l) and (m) provide the requirements for shift complement regarding licensed operators. The regulations describe the minimum shift composition for operating modes, as well as cold shutdown and refueling. Additionally, Section 5.1.2 and new Specifications 5.2.2.a, 5.2.2.b and 5 2.2.c specify when the licensed operators are required to be in the control room. Removing the Table from TS will not jeopardize plant safety nor is it necessary to be duplicated in order to assure safe operation of the facility.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.4 The requirement for an SRO to be present during fuel handling and to supervise all core alterations need not be retained in technical specifications. Duplication of the regulation provided 10 CFR 50.54(m)(2)(iv) is not necessary to assure safe operation of the facility. The current regulation states,

"Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person."

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

RELOCATION NOTES

C.5 The requirements on training may be deleted from Technical Specifications on the basis that they are adequately addressed by other Section 5.0 administrative controls as well as by regulations. Technical Specification Section 5.3, Unit Staff Qualifications, provides adequate requirements to assure an acceptable, competent operating staff. Each member of the unit staff shall meet or exceed the minimum qualifications of specific Regulatory Guides or ANSI Standards acceptable to the NRC staff. Section 5.3 of Technical Specifications describes the details of the required qualifications.

In addition, Technical Specification Section 5.2, Organization, details unit staff requirements. Specifications 5.2.2.a, 5.2.2.b, 5.2.2.c, and 10 CFR 50.54 describe the minimum shift crew composition and delineate which positions require an RO or SRO license. Training and requalification of those positions are as specified in 10 CFR 55.

Based upon these considerations, duplicating the provisions relating to training in Section 5.4 of the TS is not necessary to assure operation of the facility in a safe manner.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.6 The review and audit functions may be relocated from Technical Specifications on the basis that they can be adequately addressed elsewhere. The TS provisions are not necessary to assure safe operation of the facility, given the requirements in the Quality Assurance (QA) Program implementing 10 CFR 50.54 and 10 CFR Part 50, Appendix B to control the requirements for all review and audit functions except those associated with the security and emergency plans.

The security and emergency plan review and audit functions are relocated to their respective plans in accordance with Generic Letter 93-07. Such an approach results in an equivalent level of regulatory authority while providing for a more appropriate change control process. The net effect of the change is that the level of safety of plant operation is unaffected and NRC and facility resources associated with processing

C.6 (cont.) license amendments to this administrative control are optimized. The following points summarize the reasons for removing the review and audit requirements from standard technical specifications.

1. The on-site review function, composition, alternate membership, meeting frequency, quorum, responsibilities, authority and records are all covered in equivalent detail in ANSI N18.7-1976. These requirements are also proposed to be covered in the QA Program, FSAR, or appropriate plant procedures. Equivalent change control is provided by 10 CFR 50.54 or 10 CFR 50.59.

2. The off-site review group is also addressed, although with less detail, in ANSI N18.7-1976. The QA Program, FSAR, or appropriate plant procedures will include the requirements for the off-site review group. This organization is not necessary to assure safe operation of the facility. Relocation of the requirements from the Technical Specifications to another document with equivalent change control is appropriate.

3. Audit requirements are specified in the QA Program to satisfy 10 CFR 50, Appendix B, Criterion XVIII. Audits are also covered by ANSI N18.7, ANSI N45.2, 10 CFR 50.54(t), 10 CFR 50.54(p), and 10 CFR 73. Therefore, duplication of the requirements contained in the above documents by the Administrative Controls Section of the Technical Specifications does not enhance the level of nuclear safety for the unit. Therefore, the provisions relating to audits are not necessary to assure safe operation of the facility.

4. The requirements for record retention may be relocated from the TS on the basis that they are adequately addressed by the QA Program (10 CFR Part 50, Appendix B, Criteria XVII) and the related QA Plan.

   Facility operations are performed in accordance with approved written procedures. Areas include normal startup, operation and shutdown, abnormal conditions and emergencies, refueling, safety related maintenance, surveillance and testing, and radiation control. Facility records document appropriate station operations and activities. Retention of these records provides documentation retrievability for review of compliance with requirements and regulations. Post-compliance review of records does not directly assure operation or the facility in a safe manner, as activities described in these documents have already been performed. In addition, numerous other regulations such as 10 CFR Part 20, Subpart L, and 10 CFR 50.71 require the retention of certain records related to operation of the nuclear plant. Existing regulatory requirements provide sufficient control of record keeping provisions and removing them from TS is acceptable.

   This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.7 The markup proposes to include the bases control program under Section 5.5, Programs and Manuals. Additionally, editorial changes were made for clarifications purposes. Technical content of the bases control program remains the same.

C.8 Delete the paragraph heading for 5.7.1.1, Scope, as subsequent paragraphs have also been proposed to be deleted. Renumber 5.7 to 5.4, editorial changes consistent with markup proposal.

The markup reflects the proposal to relocate the requirements to establish, implement, and maintain procedures related to the Emergency Plan and Security Plan. Since the Security Plan requirements are specified in 10 CFR 50.54, 73.40, 73.55, and 73.56 and the Emergency Plan requirements are specified in 10 CFR 50.54 and 10 CFR Part 50, Appendix E, Section V, the staff has issued a Generic Letter (93-07) to remove the requirements from the standard technical specification and relocate them to their respective plans.

The requirements for the review of the security program and implementing procedures and for the review of the station emergency plan and implementing procedures will be included in their respective plans. Further changes in these review requirements must be made in accordance with 10 CFR 50.54(p) for the Security Plan and 10 CFR 50.54(q) for the Emergency Plan. The extensive requirements for emergency planning in 10 CFR 50.47 and 50.54 and for security in 10 CFR 50.54 and 73.55 for drills, exercises, testing, and maintenance of the program, provide adequate assurance that the objective of the previous TS for a periodic review of the program and changes to the programs will be met. Therefore, duplication of the requirements contained in the regulations does not enhance the level of nuclear safety.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.9 Review and approval process and the temporary change process for procedures are adequately addressed in TVA-NQA-PLN89-A and WBN plant procedures. This proposal is based on the existence of the following requirements which duplicate 10 CFR 50.36 in these areas.

The requirement for procedure control is mandated by 10 CFR 50, Appendix B, Criterion II and Criterion V. ANSI N18.7-1976, which is an NRC staff-endorsed document used in the development of many licensee QA plans, also contains specific requirements related to procedures. ANSI N18.7-1976, Section 5.2.2 discusses procedural adherence. This section clearly states that procedures shall be followed, and the requirements for use of procedures shall be prescribed in writing. ANSI N18.7-1976 also discussed temporary changes to procedures, and requires review and approval of procedures to be defined. ANSI N18.7-1977, Section 5.2.15 describes the review, approval and control of procedures. This section describes the requirements for the licensee's QA Program to provide measures to control and coordinate the approval and issuance of

C.9 (cont.) documents, including changes thereto, which prescribe all activities affecting quality. The section further states that each procedure shall be reviewed and approved prior to initial use. The required reviews are also described. ANSI N45.2-1971, Section 6, also specifies that the QA Program describe procedure requirements.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.10 The Radiation Protection Program has been relocated to WBN plant procedures. The Radiation Protection Program requires procedures to be prepared for personnel radiation protection consistent with the requirements of 10 CFR Part 20. The requirement to have procedures to implement Part 20 is also contained within 10 CFR 20.1101(b). Periodic review of these procedures is addressed under 10 CFR 20.1101(c). The program requirements specified above are described in the FSAR or appropriate plant procedures.

These provisions do not satisfy the policy statement guidance for inclusion in TS. The requirements of the rule provide sufficient control of these provisions, and that 10 CFR 50.59 provides adequate controls for those provisions in the FSAR and plant procedures.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.11 The Process Control Program and appropriate plant procedures implement the requirements of 10 CFR Part 20, 10 CFR Part 61, and 10 CFR Part 71. The regulatory controls stated above provide sufficient control of these requirements and removing these provisions from the TS is acceptable.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.12 The In-Plant Radiation Monitoring Program provides controls to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program was developed to minimize radiation exposure to plant personnel (post-accident). The In-Plant Radiation Monitoring Program administrative control does not involve monitoring process variables that are initial conditions for a design basis transient or accident, nor does it involve a primary success path to mitigate a DBA. Therefore, this program has been relocated to a number of different plant procedures. The training aspect is contained in the training program for personnel.

C.12 (cont.) The provisions for monitoring and performing maintenance of sampling and analysis equipment are addressed in the chemistry and radiation protection procedures.

These provisions are addressed in WBN plant procedures. The provisions do not satisfy the policy statement guidance for inclusion in TS. Therefore, the control of these provisions under 10 CFR 50.59 is acceptable.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.13 The Radiological Environmental Monitoring Program requires that procedures be prepared for monitoring the radiation and radionuclides in the environs of the plant consistent with the guidance specified in 10 CFR Part 50, Appendix I. These procedures are developed to ensure that radioactive effluents are restricted to levels as low as reasonably achievable, and have no impact on plant nuclear safety. The details and description of the program are already contained in the ODCM, as specified by existing TS 6.8.4.b. These regulatory requirements provide sufficient control of these provisions and removing them from TS is acceptable.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.14 References to in service inspection required by ASME Section XI have been eliminated, as the requirement is duplicated in 10 CFR 50.55a.

PWRs have changed the title of the Inservice Inspection Program to Reactor Coolant Pump Flywheel Inspection Program.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.15 The program description for Steam Generator Tube Surveillance has been replaced with a Reviewers's Note. The Note would require the Licensee's current steam generator tube surveillance requirements to be relocated from the LCO and included in this section. This approach was discussed with NRC by Generic Traveler NRC-14.

C.16 The Fire Protection Program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to assure the capability to achieve and maintain safe shutdown in the event of a fire. The administrative control was originally developed to ensure the capability to provide for alternate/dedicated safe shutdown in accordance with 10 CFR 50, Appendix R. As such, it allows for the ability to place the unit in a safe condition in the event of a fire.

The relocation of this administrative control from TS is also consistent with the guidance in NRC Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements." In that letter, the staff concluded that the provisions of 10 CFR 50.59 should apply directly to changes the licensee desired to make in the fire protection program so long as those changes did not adversely affect the ability to achieve and maintain safe shutdown. The license condition described in GL 86-10., stated that changes which adversely affect the ability to achieve and maintain safe shutdown in the event of a fire require prior approval of the staff. Thus, the license condition established as part of the NRC GL 86-10 implementation also makes this administrative control unnecessary. Therefore, the control of these provisions under the terms of the license conditions and 10 CFR 50.59 is acceptable.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.17 This section has been renumbered as 5.7.2.18. The Safety Function Determination Program is now included in ITS under the section "Programs". The generic example has been removed from STS.

C.18 The requirement to submit a Startup Report has been deleted from the ITS. The report was a summary of plant startup and power escalation testing following receipt of the Operating License, an increase in licensed power level, the installation of nuclear fuel with a different design or manufacturer than the current fuel, and modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report provided a mechanism for the staff to review the appropriateness of licensee activities after-the-fact, but contained no requirement for staff approval.

The approved 10 CFR Part 50, Appendix B, QA Plan and Startup Testing Program (FSAR) provide assurance that the listed activities are adequately performed and that appropriate corrective actions, if required, are taken.

Inasmuch as this report was required to be provided to staff within 90 days following completion of the respective milestone, it was clearly not necessary to assure operation of the facility in a safe manner for the interval between completion of the startup testing and submittal of the report. Additionally, because there was no requirement for the staff to approve the report, the Startup Report is not necessary to assure operation of the facility in a safe manner.

C.18 (cont.) Therefore, the removal of this requirement is considered acceptable.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.19 The proposed changes to the administrative control for the Pressure Temperature Limits Report retain those provisions that are necessary to ensure appropriate control of the pressure and temperature limits calculations, and present other guidance information in reviewers' notes. The reviewers' notes are appropriate for the STS because this is a new practice; these considerations are useful to the NRC staff and licensees that propose to adopt this provision. These changes were discussed in a meeting between EMCB and WOG on 11/22/93.

C.20 The TS requirement to submit a special report for ECCS actuations is sufficiently controlled by other regulatory requirements and removing them from TS is acceptable. 10 CFR 50.73(a)(2)(iv) provides requirements for the licensee to submit a Licensee Event Report in the event of an ECCS actuation. The report is required to be submitted within 30 days and will contain the same type of information as the special report.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.21 The requirements on record retention be relocated from the STS on the basis that they are adequately addressed by the QA Program (10 CFR Part 50, Appendix B, Criteria XVII) and the related QA Plan.

Facility operations are performed in accordance with approved written procedures. Areas include normal startup, operation and shutdown, abnormal conditions and emergencies, refueling, safety-related maintenance, surveillance and testing, and radiation control. Facility records document appropriate station operations and activities. Retention of these records provides documentation retrievability for review of compliance with requirements and regulations. Post-compliance review of records does not directly assure operation of the facility in a safe manner, as activities described in these documents have already been performed. In addition, numerous other regulations such as 10 CFR Part 20, Subpart L, and 10 CFR 50.71 require the retention of certain records related to operation of the nuclear plant. Therefore, it can be concluded that these regulatory requirements provide sufficient control of these record keeping provisions and removing them from TS is acceptable.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.22 The High Radiation Area administrative control provides access controls for high radiation areas in lieu of those specified in 10 CFR 20.203(c) [10 CFR 20.1601(a) in the ''new'' Part 20]. The controls are developed to ensure nuclear plant personnel safety and have no impact on nuclear safety. Additionally, nuclear plant personnel are not "members of the public." Thus, the principal operative standard in Section 182a of the Atomic Energy Act; "health and safety of the public" does not apply. Based on these considerations, the Radiation Protection Program administrative control is not necessary to assure operation of the facility in a safe manner and can be deleted from TS. 10 CFR 20.203(c)(5) [10 CFR 20.1601(c) in the "new" Part 20] provides the appropriate mechanism for licensees wishing to propose alternative controls to those specified in 10 CFR 20. Licensees relocating the high radiation control description currently contained in the Administrative Controls Section of Technical Specifications to another controlled document with change controls is considered acceptable. Subsequent changes to these controls will be processed in accordance with the applicable regulation.

This proposal is consistent with the guidance provided in the NRC letter dated October 25, 1993.

C.23 Other Sections of STS which referenced Sections in 5.0 have been updated to reflect new numbering.

ENCLOSURE 3

ADDITIONAL TECHNICAL SPECIFICATION CHANGES

Additional Technical Specification Changes

Additional changes being requested are identified below by the site change tracking number. The pages following the change descriptions indicate the change tracking number in addition to the proposed changes indicated in shading. Clean pages for inclusion in the "Final Draft" are provided in Enclosure 4.

Change Package 94-004

As described in the Technical Specification Bases, the RCS Flow Indicators are calibrated based on the flow measurement using a precision secondary calorimetric. This calorimetric flow measurement is performed after power exceeds 90% of rated thermal power. Prior to the initial performance of this measurement, the flow indicators are adjusted based on engineering calculations and empirical data from similar plants.

Prior to Preoperational Hot Functional testing, the flow transmitters are calibrated based on engineering equations that relate elbow tap differential pressure to flow. Then, the flow transmitters were normalized to indicate 108% for the differential pressures measured during Watts Bar's Hot Functional Testing. After core load, the flow instrument loops are renormalized to indicate 100%. The new differential pressures are used to recalculate flow based on the engineering equations and the results are acceptable if greater than 90% of the thermal design flow.

Until reactor power is raised to 90%, the only bases for judging RCS flow meets surveillance requirement SR 3.4.1.3 is that the flow indicators are indicating 100% flow which does not necessarily mean that flow is greater than the flow stated in the surveillance. Therefore, the performance of the SR 3.4.1.3 prior to performance of SR 3.4.1.4 only provides reasonable assurance that RCS flow is greater than 90% thermal design flow.

Change Package 94-030

The proposed change is necessary to provide the acceptable value for ABSCE boundary inleakage during periodic testing. The maximum allowable building inleakage rate is 6300 cfm.

Change Package 94-032

The proposed change is necessary to accurately reflect the correct reference for acceptance criteria for SR 3.6.10.3.

Change Package 94-033

The proposed change is necessary to be consistent with current design response time requirements.

Additional Technical Specification Changes

Change Package 94-039

The proposed change is necessary to be consistent with current design response time requirements.

Change Package 95-005

The proposed change is necessary to be consistent with current design calculations for operation with inoperable MSSVs.

Change Package 95-009

The proposed change is necessary to be consistent with current design requirements for normal and abnormal temperature limits in the identified areas.

Change Package 95-011

The surveillance cannot be performed until after the trip breakers are closed (part of the applicability). The proposed change is necessary to provide time to perform the initial surveillance after entering the applicability.

Change Package 95-012

Valves that are closed in the flow path with power available for opening the valve is considered adequate to satisfy this surveillance.

Change Package 95-013

The proposed change is necessary to be consistent with current design requirements for instrument accuracy of the source range nuclear instrumentation.

Change Package 95-018

The site surveillance instructions currently being written need the proposed change to more clearly identify what instrumentation and control functions are required to be tested.

Change Package 95-019

Valves that are closed in the flow path with power available for opening the valve is considered adequate to satisfy this surveillance.

Additional Technical Specification Changes

Change Package 95-021

Testing of the starting air system during the preoperational test program confirmed that 5 consecutive starts can be made with an initial pressure of 190 psig. At 170 psig, at least 3 or 4 starts can be made. This lower pressure (170 psig) can be used as a lower limit to assure that the diesel engine will start.

Change Package 95-022 through 95-026

The proposed changes are as a result of additional certification comments from Westinghouse.

Change Package 95-027

The criticality analysis for the Watts Bar Nuclear Plant Spent Fuel Storage Racks Accident Analysis section assumes normal pool conditions including 2000 ppm soluble boron concentration at accident initiation.

Change Package 95-028

Change is required based on recent engineering calculations for this instrument.

Change Package 95-030

The 37 test plan has been clarified and the 55 plan has been abandoned as issued in subsection ISTD of ASME OM Code-1995. A combination of the 37 plan and the 10% plan is best suited for WBN's population of snubbers (i.e. ≈ 900 mechanical snubbers and 20 hydraulic snubbers). See App. D of ISTD for comparison of the plans.

Change Package 95-032

The Bases explain that this test must be performed in MODE 3 to establish testing conditions to meet the acceptance criteria. The test cannot be performed in MODES 1 or 2 because this would create an unwanted plant trip.

Change Package 95-034

The proposed change is necessary to be consistent with current design response time requirements.

Additional Technical Specification Changes

Change Package 95-035

The frequency for cycling the Reactor Head Vent System block and flow control valves is being changed to be in accordance with the Inservice Testing Program alternate test frequency. This change is being made because industry experience has indicated that Target Rock valves, such as those installed at WBN, sometimes do not fully reseat after cycling. These valves should be cycled during cold shutdown to preclude the possibility of excessive Reactor Coolant System leakage through these valves at power in the event that the valves do not fully reseat after being cycled.

The proposed alternate test frequency of once per cold shutdown, not to exceed once per 92 days during frequent cold shutdowns, was submitted to the NRC in the Inservice Test (IST) Program dated July 22, 1994, and the NRC generically approved all of the alternate frequency justifications in the IST Program in NRC Safety Evaluation Report (NUREG-0847), Supplement 14.

Change Package 95-037

The proposed change is necessary to clarify the performance frequency of the surveillance.

Change Package 95-040

The deleted valves will be maintained in a locked position.

Change Package 95-041

The proposed change is necessary to be consistent with current allowable inleakage requirements on this system.

Change Package 95-042

The proposed change is needed to clarify the actions and completion time. As written, the action to "Be in MODE 3" could be misunderstood to mean if in MODE 4, go to MODE 3. The completion time as written would require completing the Stage II plan 17 hours prior to predicted flooding when it should only require completion within 17 hours or prior to the predicted flooding which could be much longer than 17 hours.

Additional Technical Specification Changes

Change Package 95-043

The proposed change is necessary to be consistent with current Appendix R program.

Change Package 95-045

The proposed change is necessary to be consistent with current battery type used at WBN Unit 1.

Change Package 94-019

The proposed change is necessary to be consistent with current design requirements for the Reactor Trip System and ESF System.

Editorial Changes for the following pages:

3.3-38; 3.4-27; B 3.3-117; B 3.3-168; B 3.8-68; B 3.8-102; B 3.9-1; TR B 3.1-2; TR B 3.9-3; 5.0-4; TR 3.4-5; TR B 3.4-9

CP 94-004

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ 2214 psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ≤ 593.5°F.	12 hours
SR 3.4.1.3	-------------------NOTE------------------- Not required to be performed until 12 hours after the installed flow instrumentation has been calibrated to the precision heat balance of SR 3.4.1.4. -- Verify RCS total flow rate is ≥ 397,000 gpm (process computer) or 401,000 gpm (control board indication).	12 hours
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 24 hours after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 397,000 gpm.	18 months

CP 94-004

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

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

This SR is modified by a Note which permits delaying performance of the SR on the initial reactor startup until after the installed flow instrumentation has been calibrated to the data obtained in SR 3.4.1.4.

SR 3.4.1.4

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

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

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis," Section 15.2, "Normal Operation and Anticipated Transients," and Section 15.3.4, "Complete Loss Of Forced Reactor Coolant Flow."

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two ABGTS trains inoperable during movement of irradiated fuel assemblies in the fuel handling area.	D.1 Suspend movement of irradiated fuel assemblies in the fuel handling area.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABGTS train for $\geq$ 10 continuous hours with the heaters operating.	31 days
SR 3.7.12.2	Perform required ABGTS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.12.4	Verify one ABGTS train can maintain a pressure between -0.25 and -0.5 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate $\geq$ 8100 and $\leq$ 9900 cfm. During this Surveillance, the ABSCE boundary inleakage shall not exceed 6300 cfm.	18 months on a STAGGERED TEST BASIS

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 8.

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 and -0.5 inches water gauge with respect to atmospheric pressure at a nominal flow rate $\geq$ 8100 and $\leq$ 9900 cfm. During this periodic testing, the ABSCE boundary inleakege rate shall not exceed 6300 cfm. Periodic testing of ABGTS shall be performed once with one AB general supply fan running and one train of its discharge ABSCE isolation dampers open and once with one purge air supply fan running and one of its suction-side ABSCE isolation dampers open. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 9).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 8.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, "Engineered Safety Feature (ESF) Filter Systems."

2. Watts Bar FSAR, Section 9.4.2, "Fuel Handling Area Ventilation System."

3. Watts Bar FSAR, Section 15.0, "Accident Analysis."

4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."

(continued)

CP 94-032

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1 (continued)

excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.10.2

Verifying ARS fan motor current with the return air backdraft dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.10.3

Verifying the OPERABILITY of the air return damper to the proper opening torque (Ref. 3) provides assurance that the proper flow path will exist when the fan is started. By applying the correct torque to the damper shaft, the damper operation can be confirmed. The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.

REFERENCES

1. Watts Bar FSAR, Section 6.8, "Air Return Fans."
2. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."
3. System Description N3-30RB-4002.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.9.3	Verify each EGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.6.9.4 CP 94-033	Verify each EGTS train produces a flow rate ≥ 3600 and ≤ 4400 cfm within 20 seconds from the initiation of a Containment Isolation Phase A signal.	18 months on a STAGGERED TEST BASIS

BASES

BACKGROUND (continued)

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters are included to reduce the relative humidity of the airstream on systems that operate in high humidity. Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and adsorbers. Cross-over flow ducts are provided between the two trains to allow the active train to draw air through the inactive train and cool the air to keep the charcoal beds on the inactive train from becoming too hot due to absorption of fission products.

The containment annulus vacuum fans maintain the annulus at -5 inches water gauge vacuum during normal operations. During accident conditions, the containment annulus vacuum fans are isolated from the air cleanup portion of the system.

The EGTS reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the EGTS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The EGTS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the EGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA.

CP- 94-033

The safety analysis assumes an initial annulus vacuum pressure of -5.0 inches water gauge prior to the LOCA. The analysis further assumes that upon receipt of a Phase A isolation signal from the RPS, the EGTS fans automatically start and achieve a minimum flow of 3600 cfm

(continued)

CP 94-033

TR 3.3 INSTRUMENTATION

TR 3.3.2 Engineered Safety Features Actuation System Instrumentation

TR 3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks as shown in Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5, LCO 3.3.5; Technical Specification 3.3.6, Table 3.3.6-1; and Technical Specification 3.6.9 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3.2-1 of this document.

APPLICABILITY: As shown in Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5 Applicability; and Technical Specification 3.3.6 Applicability; and Technical Specification 3.6.9 Applicability.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refer to Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5 Actions; Technical Specification 3.3.6 Actions; and Technical Specification 3.6.9 Actions.	A.1 Refer to Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5 Actions; Technical Specification 3.3.6 Actions; and Technical Specification 3.6.9 Actions.	Refer to Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5 Actions; Technical Specification 3.3.6 Actions; and Technical Specification 3.6.9 Actions.

Table 3.3.2-1 (Page 1 of 5)

Engineered Safety Features Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Ventilation Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Raw Cooling Water	N.A.
j. CREVS Actuation	N.A.
k. Containment Air Return Fan	N.A.
l. Component Cooling System	N.A.
m. Start Diesel Generators	N.A.
n. Reactor Trip	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 27(4)/32(14)/37(5)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8(3)
3) Containment Isolation-Phase "A"(6)	≤ 12(2)/22(1)
4) Containment Ventilation Isolation	≤ 5.0(2)(11)
5) Auxiliary Feedwater Pumps	≤ 60(10)
6) Essential Raw Cooling Water	≤ 47(2)/57(1)
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50(2)/60(1)
9) Start Diesel Generators	≤ 12(12)
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27(4)/32(14)/37(5)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8(3)
3) Containment Isolation-Phase "A"(6)	≤ 12(2)/22(1)
4) Containment Ventilation Isolation	≤ 5.0(2)(11)
5) Auxiliary Feedwater Pumps	≤ 60(10)

CP 94-039

(continued)

CR 94-039

Table 3.3.2-1 (Page 2 of 5)

Engineered Safety Features Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low (continued)	
6) Essential Raw Cooling Water	≤ 47[2]/57[1]
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50[2]/60[1]
9) Start Diesel Generators	≤ 12[12]
4. Steam Line Pressure Negative Rate-High	
a. Steam Line Isolation	≤ 7
5. Steam Line Pressure - Low	
a. Safety Injection (ECCS)	≤ 27[4]/32[14]/37[5]
1) Reactor Trip (from SI)	≤ 2
2) Feedwater Isolation	≤ 8[3]
3) Containment Isolation-Phase "A"[6]	≤ 12[2]/22[1]
4) Containment Ventilation Isolation	≤ 5.0[2][11]
5) Auxiliary Feedwater Pumps	≤ 60[10]
6) Essential Raw Cooling Water	≤ 47[2]/57[1]
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50[2]/60[1]
9) Start Diesel Generators	≤ 12[12]
b. Steam Line Isolation	≤ 7
6. Containment Pressure - High - High	
~~a. Containment Spray~~	≤ 221[13]
b. Containment Isolation-Phase "B"	≤ 68[2]/78[1]
c. Steam Line Isolation	≤ 7
d. Containment Air Return Fans	480 ≤ RT ≤ 600
7. Steam Generator Water Level - High - High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 8[3]

(continued)

Table 3.3.2-1 (Page 3 of 5) CP 94-039

Engineered Safety Features Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Steam Generator Water Level - Low - Low Coincident with Vessel ΔT $\leq$ 50% RTP	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq 60^{(7)}$
b. Turbine-driven Auxiliary Feedwater Pumps	$\leq 60^{(8)}$
9. Steam Generator Water Level-Low-Low Coincident with Vessel ΔT > 50% RTP	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq 60^{(7)}$
b. Turbine-driven Auxiliary Feedwater Pumps	$\leq 60^{(8)}$
10. RWST Level-Low Coincident with Containment Sump Level-High and Safety Injection	
Automatic Switchover to Containment Sump	$\leq$ 250
11. Loss-of-Offsite Power	
Auxiliary Feedwater Pumps	$\leq$ 60
12. Trip of All Main Feedwater Pumps	
Auxiliary Feedwater Pumps	$\leq$ 60
13. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq$ 40
b. Turbine-driven Auxiliary Feedwater Pumps	$\leq$ 38
14. Loss of Voltage/Degraded Voltage	
6.9 kV Shutdown Board	$\leq 12^{(9)}$
15. MSV Vault Room Water Level - High	
a. North MSV Vault Room	$\leq 8.5^{(15)}/12^{(16)}$
b. South MSV Vault Room	$\leq 8.5^{(15)}/12^{(16)}$

Table 3.3.2-1 (Page 4 of 5) CP 94-039

Engineered Safety Features Response Times

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air operated valves.

(4) Offsite power available - diesel generator starting and sequence loading delays not included. Response time limit includes the opening of valves to establish flowpath and bringing the pumps to full speed. The additional sequential transfer of CCP suction from the VCT to the RWST (RWST valves open, then the VCT valves close) is included.

(5) Diesel generator starting and sequence loading delays included. Response time limit includes the opening of valves to establish flow path and bringing the pumps up to full speed. The additional sequential transfer of suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(6) The following equipment are exceptions to the response time shown in the table and will have the following response times for the initiating signals and functions:

A. Fire Protection CIVs	22[(2)]/32[(1)]
B. Ice Condenser CIVs	32
C. Excess Letdown Hx Supply CIV	68[(2)]/78[(1)]
D. EGTS Fans	20[(2)]/30[(1)]
E. Required for EGTS OPERABILITY	
1. Fire Protection Secondary CIVs	20[(2)]/30[(1)]
2. Secondary Containment Purge Isolation Valves	12.7[(2)]/22.7[(1)]

(7) On 2/3 any steam generator and Trip Time Delay = 0 seconds.

(8) On 2/3 in 2/4 steam generators and Trip Time Delay = 0 seconds.

(9) The response time is measured from the time the 6.9 kV shutdown boards voltage exceeds the Setpoint until the time full voltage is returned for the loss of voltage sensors; or from the time the degraded voltage timers generate a signal to trip the feeder breakers and shed loads until the time full voltage is returned for the degraded voltage sensors.

(10) The Response Time for motor-driven AFW pumps includes the diesel generator starting and sequence loading delays. The Response Time for (steam) turbine driven AFW pumps does not include diesel generator starting and sequence loading delays.

(continued)

CP-94-039

Table 3.3.2-1 (Page 5 of 5)

Engineered Safety Features Response Times

TABLE NOTATIONS

(11) Containment purge valves only. Containment radiation monitor valves have a response time of 6.5 seconds.

(12) Diesel generator start time includes a reactor trip response time of 2 seconds.

(13) Includes diesel generator starting, containment spray pump sequence loading-delay/breaker closure, plus stroke time of 1-FCV-72-39/2.

(14) Diesel generator starting and sequence loading delays included. Response time limit includes the opening of valves to establish flowpath and bring pumps to full speed. The additional sequential transfer of ECCS pump suction from the VCT to the RWST (RWST valves open) is included.

(15) Feedwater Isolation Valve (motor) and Feedwater Regulating Valve (air operated) response time includes an ESFAS signal response time of 2 seconds.

(16) Feedwater pumps coast down to zero flow response time includes an ESFAS signal response time of 2 seconds.

CP 95-005

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Applicable Power in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	APPLICABLE POWER (% RTP)
5	≤ 100
4	≤ 59
3	≤ 42
2	≤ 26

CP 95-005

BASES

LCO (continued)

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1 above 40% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below 40% RTP in MODES 1, 2, and 3, only two MSSVs per steam generator are required to be OPERABLE.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

With one or more MSSVs inoperable, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. With an upper MTC limit of 0 $\Delta k/k/°F$, this is accomplished by maintaining THERMAL POWER at or below the power levels specified in Table 3.7.1.1. This ensures that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. The reduced THERMAL POWER level for a reduced steam relieving capacity can be determined by performing a energy balance between the reactor coolant system heat

(continued)

BASES

ACTIONS

A.1 (continued)

generation and the steam relief through the OPERABLE MSSVs, as shown below:

$$\text{Allowable THERMAL POWER Level } (\%) = 100\frac{4w_s h_{fg}}{QK}$$

where: w_s = Minimum total steam relief capacity of the OPERABLE MSSVs on any one steam generator, in lbm/sec.

h_{fg} = heat of vaporization at the highest MSSV full-open pressure, in Btu/lbm.

Q = NSSS power rating of the plant (includes reactor coolant pump heat) in MWt.

K = Unit conversion factor: 947.82 Btu/sec/MWt.

Note: The values in Specification 3.7.1 include an allowance for instrument and channel uncertainties to the allowable RTP obtained with this algorithim.

(continued)

CP 95-009.

LIST OF TABLES

LIST OF EFFECTIVE PAGES

CP 95-009

TR 3.7 PLANT SYSTEMS

TR 3.7.5 Area Temperature Monitoring

TR 3.7.5 The normal temperature limit of each area shown in Table 3.7.5-1 shall not be exceeded for > 8 hours and the abnormal temperature limits shall not be exceeded.

APPLICABILITY: Whenever the affected equipment in an area is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more areas exceeding normal temperature limits for > 8 hours.	------------NOTE------------ TR 3.0.3 and TR 3.0.4 are not applicable. ---------------------------- A.1 Prepare and submit to the NRC a report in accordance with 10 CFR 50.4 that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	30 days

(continued)

CP 95-009

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more areas exceeding abnormal temperature limits except for the IPS Mechanical or Electrical Equipment Rooms (Areas 31, 32, or 34). OR Mechanical or Electrical Equipment Rooms (Areas 23, 25, 35, or 36) temperature less than 35°F.	B.1.1 Restore the area(s) to within normal temperature limits.	4 hours
	OR	
	B.1.2 Declare the affected equipment in the affected area(s) inoperable.	4 hours
	AND	
	B.2 Prepare and submit to the NRC a report in accordance with 10 CFR 50.4 that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	30 days
C. Mechanical or Electrical Equipment Rooms in Intake Pumping Station (Areas 31, 32, or 34) less than 40 °F and greater than 32 °F.	C.1 Initiate action to maintain temperatures greater than 32 °F.	24 hours
	AND	
	C.2 Restore temperatures to within normal limits.	7 days

CP 95-009

D. Mechanical or Electrical Equipment Rooms in Intake Pumping Station (Areas 31, 32, or 34) 32 °F or less.	D.1 Declare the affected equipment in the affected area(s) inoperable. AND D.2 Prepare and submit to the NRC a report in accordance with 10 CFR 50.4 that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	Immediately 30 days

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.5.1	Verify each area temperature is within limits.	12 hours

CP 95-009

Table 3.7.5-1
Area Temperature Monitoring

AREA	NORMAL LIMIT °F	ABNORMAL LIMIT °F
1. Aux Bldg el 772 next to 480V Sd Bd transformer 1A2-A.	≤ 104	≤ 110
2. Aux Bldg el 772 next to 480V Sd Bd transformer 1B1-B.	≤ 104	≤ 110
3. Aux Bldg el 772 next to 480V Sd Bd transformer 2A2-A.	≤ 104	≤ 110
4. Aux Bldg el 772 next to 480V Sd Bd transformer 2B2-B.	≤ 104	≤ 110
5. Aux Bldg el 772 next to 480V Rx MOV Bd 1A2-A.	≤ 83	≤ 104
6. Aux Bldg el 772 next to 480V Rx MOV Bd 2A2-A.	≤ 83	≤ 104
7. Aux Bldg el 772 next to 480V Rx MOV Bd 2B2-B.	≤ 83	≤ 104
8. Aux Bldg el 772 across from spare 125V vital battery charger 1-S.	≤ 83	≤ 104
9. Aux Bldg el 772 U1 Mech Equip Room.	≤ 91	≤ 104
10. Aux Bldg el 757 U1 Sd Bd room behind stairs S-A3.	≤ 85	≤ 104
11. Aux Bldg el 757 U2 Sd Bd room behind stairs S-A13.	≤ 85	≤ 104
12. Aux Bldg el 757 U1 Refueling beside Aux boration makeup tk.	≤ 104	≤ 115
13. Aux Bldg el 737 U1 outside supply fan room.	≤ 104	≤ 110
14. Aux Bldg el 713 U1 across from AFW pumps.	≤ 104	≤ 110
15. Aux Bldg el 692 U1 outside AFW pump room door.	≤ 104	≤ 110
16. Aux Bldg el 692 U2 near boric acid concentrate filter vault.	≤ 104	≤ 110
17. Aux Bldg el 676 next to 0-L-629.	≤ 104	≤ 110
18. North steam valve vault room U1. (at MSSVs)	≥ 80	≥ 80

(continued)

CP 95-009

Table 3.7.5-1
Area Temperature Monitoring

AREA	NORMAL LIMIT °F	ABNORMAL LIMIT °F
19. South steam valve vault room U1. (at MSSVs)	≥ 80	≥ 80
20. Add Equip Bldg U1 el 729 between UHI accumulators.	≤ 92	≤ 110
21. CB Main Control Room south wall.	≤ 80	≤ 104
22. CB Main Control Room across from 1-M-9.	≤ 80	≤ 104
23. CB Computer room el 708 center of room.	≤ 74	≤ 104
24. CB Aux. Instrument Room el 708.	≤ 90	≤ 104
25. D/G Bldg el 742 2B-B D/G room on wall by battery charger.	≤ 104	≤ 120
26. D/G Bldg el 742 1A-A D/G Room near D/G set.	≥ 50	≥ 50
27. D/G Bldg el 742 1B-B D/G Room near D/G set.	≥ 50	≥ 50
28. D/G Bldg el 742 2A-A D/G Room near D/G set.	≥ 50	≥ 50
29. D/G Bldg el 742 2B-B D/G Room near D/G set.	≥ 50	≥ 50
30. D/G Bldg el 760.5 next to 480V diesel Aux Bd 2B1-B.	≤ 104	≤ 120
31. IPS Mechanical Equipment Room 1 el 722 near ERCW and HPFP Instruments and sense lines.	≥50≤104	≥40≤115
32. IPS Mechanical Equipment Room 2 el 722 near ERCW and HPFP Instruments and sense lines.	≥50≤104	≥40≤115
33. IPS el 741 in B train ERCW pump room.	≤ 120	≤ 120
34. IPS el 711 next to 480V IPS board and transformer (A bus).	≥50≤104	≥40≤115
35. IPS el 711 next to 480V IPS board and transformer (B bus).	≤ 104	≤ 115
36. Add D/G Bldg el 742 C-S D/G Room near D/G set.	≥ 50	≥ 50

CP 95-009

BASES

BACKGROUND (continued)	The general guidelines, which are followed for the qualification of electrical equipment, are provided in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 1). Detailed requirements for the implementation of the general guidelines are provided in various Regulatory Guides and IEEE Standards. Basic requirements for the qualification of mechanical equipment are outlined in General Design Criteria 4 (Ref. 2).
APPLICABLE SAFETY ANALYSIS	Certain components, which have the service temperatures controlled by this requirement, are part of the primary success path and function to mitigate DBAs and Transients. However, the integrity/OPERABILITY of these components is addressed in the relevant specifications that cover individual components. The service temperatures and the thermal aging, which are controlled by observing the requirements of this TR, are not inputs to the safety analysis. Further, Probabilistic Risk Assessment studies performed to date, do not explicitly model the function of area temperature monitors. In addition, this particular requirement covers only service temperatures and thermal aging of these components, which are not considerations in designing the accident sequences for theoretical hazard evaluations (Ref. 3).
TR	TR 3.7.5 provides nominal temperature limits in the vicinity of major equipment. The TR allows for each area shown in Table 3.7.5-1 to be higher or lower than the normal limit for a maximum of eight hours.
APPLICABILITY	The limits on temperature and time apply whenever the affected equipment in an affected area is required to be OPERABLE.

(continued)

CP 95-009

BASES (continued)

ACTIONS

A.1

Whenever the temperature in one or more areas have exceeded the normal temperature limits for more than eight hours, a report must be prepared and submitted to the NRC within 30 days. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation, and prepare the report.

Condition A has been modified by a Note stating that the provisions of TR 3.0.3 and TR 3.0.4 do not apply.

B.1.1, B.1.2, and B.2

Whenever the temperature in one or more areas exceed the abnormal temperature limits, the temperature must be restored to within the normal limits in 4 hours. The Completion Time of 4 hours is based on operator experience and is a reasonable time for restoring the temperature. Alternatively, the affected equipment must be declared inoperable. Within 30 days a report must be prepared and submitted to NRC. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation and prepare the report.

C.1 and C.2

Whenever the temperature in the Intake Pumping Station mechanical or electrical equipment rooms exceeds the lower limit of 40 °F, actions must be initiated within 24 hours to ensure the temperature does not decrease below 32 °F. The Completion Time of 24 hours is based on temperature analysis. Within 7 days, restore normal temperatures within the areas affected. The 7 day Completion Time is based on a reasonable repair duration, and compensatory actions available during the interim period to maintain temperatures above 32 °F.

(continued)

CP 95-009

BASES

ACTIONS (continued)

D.1 and D.2

If the temperature in the Intake Pumping Station mechanical or electrical equipment rooms decrease to 32 °F or lower, the affected equipment must be immediately declared inoperable. The Completion Time is based on potential freezing of safety-related components. Within 30 days a report must be prepared and submitted to NRC. The report must contain the cumulative time and amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation and prepare the report.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.5.1

The temperature in each area must be determined every 12 hours to ensure compliance with the limits. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

REFERENCES

1. 10 CFR 50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants".

2. 10 CFR 50 Appendix A, General Design Criteria 4, "Environmental and Dynamic Effects Design Bases".

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

CP 95-011

TR 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.7 Position Indication System, Shutdown

TR 3.1.7 The group demand position indicators shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown or control rod that is not fully inserted.

APPLICABILITY: MODES 3, 4, and 5, when the reactor trip breakers are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more group demand position indicators inoperable.	A.1 Open reactor trip breakers.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.7.1	Determine that each group demand position indicator is OPERABLE by movement of the associated shutdown or control rod 10 steps in any one direction.	Within 4 hours after closing the reactor trip breakers AND 31 days thereafter

CP 95-011

BASES (continued)

ACTIONS

A.1

With one or more group demand position indicators inoperable, the plant must be placed in a condition where the demand position indicators are not required. This is accomplished by opening the reactor trip breakers immediately.

The immediate Completion Time is consistent with the required time for actions to be pursued without delay and in a controlled manner.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.7.1

Exercising rods at a Frequency of 31 days allows the operator to determine that all withdrawn rods, including the group step counter demand position indicator, continue to be OPERABLE. A movement of 10 steps is adequate to demonstrate motion and verify a corresponding step change in the group step counter demand position indicator. Four hours is provided to perform the first surveillance after closing the reactor trip breakers. The 31-day Frequency takes into consideration other information available to the operator in the control room and the remote likelihood that rods would be withdrawn from fully inserted for extended periods of time during shutdown conditions.

REFERENCES

1. Watts Bar FSAR, Section 4.2.3 "Reactivity Control System."

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

CP 95-012

BASES

ACTIONS

C.1 and C.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

SR 3.6.6.2

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 4). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

CP 95-012

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

acid storage tanks is required OPERABLE. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

TSR 3.1.1.2

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

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

REFERENCES

1. Watts Bar FSAR, Section 9.3.4, "Chemical and Volume Control System."

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

3. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar Nuclear Plant, Unit 1, Revision 00, April 1993.

CP 95-012

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.2.1

This surveillance verifies the temperature of the required flow path from the boric acid tanks to be at least 63°F. This ensures that the high concentration of boric acid in the storage tanks is not allowed to precipitate due to cooling.

The surveillance is modified by a note stating that the surveillance is required only if the flow path from the boric acid storage tanks is used as one of the two required flow paths. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

TSR 3.1.2.2

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

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

TSR 3.1.2.3

This surveillance demonstrates that each automatic valve in the flow path actuates to its required position on an actual or simulated actuation signal. The 18-month Frequency was developed considering it is prudent that this surveillance only be performed during a plant outage. This is due to the plant conditions needed to perform the TSR and the potential for unplanned plant transients if the TSR is performed with the reactor at power.

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

CP- 95-013

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 Neutron Monitoring System (NMS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed primary source range neutron flux monitors are fission chambers. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) with an instrument accuracy of 5% of the countrate. The detectors also provide continuous visual indication in the control room and an audible alarm to alert operators to a possible dilution accident. The NMS 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."

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

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)

CP 95-018

Table 3.3.4-1 (page 1 of 1)
Remote Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
1. Reactivity Control	
a. Source Range Neutron Flux	1
b. Reactor Trip Breaker Position	1 per trip breaker
2. Reactor Coolant System (RCS) Pressure Control	
a. Pressurizer Pressure or RCS Wide Range Pressure	1
b. Pressurizer Power Operated Relief Valve (PORV) Control or Pressurizer Block Valve Control	1 per relief path
c. Pressurizer Heater Control	1
3. RCS Inventory Control	
a. Pressurizer Level Indication	1
b. Charging Flow Control	1
c. Reactor Vessel Head Vent Valves Control or PORVs and Block Valve Control	2 per flow path
4. Decay Heat Removal via Steam Generators (SGs)	
a. RCS Hot Leg Temperature Indication	1 per loop
b. AFW Controls	1
c. SG Pressure Indication and Control	1 per SG
d. SG Level Indication or AFW Flow Indication	1 per SG
5. Decay Heat Removal via RHR System	
a. RHR FLow Control	1
b. RHR Temperature Indication	1

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown System

CP 95-018

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves or the SG atmospheric dump valves (ADVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

If the control room becomes inaccessible, the operators can establish control in the auxiliary control room, and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located in the auxiliary control room. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the remote shutdown control and instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible. Should it be necessary to go to MODE 4, decay heat removal via the Residual Heat Removal (RHR) System is available to support the transition.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

(continued)

CP 95-018

BASES

APPLICABLE SAFETY ANALYSES (continued)

The criteria governing the design and specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The Remote Shutdown System is considered an important contributor to the reduction of unit risk to accidents and as such it has been retained in the Technical Specifications as indicated in the NRC Policy Statement.

LCO

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

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

- Core reactivity control (initial and long term);
- RCS pressure control;
- Decay heat removal via the AFW System and the SG safety valves or SG ADVs;
- RCS inventory control via charging flow and Reactor Vessel Head Vent Valves or Pressurizer PORV and Block Valve operation;
- Decay Heat Removal via RHR System;
- Safety support systems though not specifically listed in Table 3.3.4-1, for the above Functions, including service water, component cooling water, reactor containment fan cooler units, auxiliary control air compressors, and onsite power, including the diesel generators are required as discussed in FSAR Section 7.4 (Reference 2).

(continued)

CP 95-019

ACTIONS

B.1 and B.2 (continued)

achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the ERCW System components or systems may render those components inoperable, but does not affect the OPERABILITY of the ERCW System.

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

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

SR 3.7.8.2

This SR verifies proper automatic operation of the ERCW System valves on an actual or simulated actuation signal. The ERCW System is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on

(continued)

CP 95-021

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with starting air receiver pressure < 190 psig and ≥ 170 psig.	E.1 Restore starting air receiver pressure to ≥ 190 psig.	48 hours
F. Required Action and associated Completion Time not met. OR One or more DGs diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each 7 day fuel oil storage tank contains ≥ 56,754 gal of fuel.	31 days

(continued)

CP 95-021

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.2	Verify lubricating oil inventory is ≥ 287 gal per engine.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG air start receiver pressure is ≥ 190 psig.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each of the four interconnected tanks which constitute the 7 day fuel oil storage tank.	31 days
SR 3.8.3.6	Perform a visual inspection for leaks in the exposed fuel oil system piping while the DG is running.	18 months
SR 3.8.3.7	For each of the four interconnected tanks which constitute the 7 day fuel oil storage tank: a. Drain the fuel oil; b. Remove the sediment; and c. Clean the tank.	10 years

CP 95-021

BASES

ACTIONS

D.1 (continued)

restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

With starting air receiver pressure < 190 psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is > 170 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit of ≥ 190 psig. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil, lube oil or starting air subsystem not within limits for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The 7 day period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

(continued)

CP 95-022

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
O. One Low Fluid Oil Pressure Turbine Trip channel inoperable.	------------NOTE------------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------	
	O.1 Place channel in trip.	6 hours
	OR	
	O.2 Reduce THERMAL POWER to < P-9.	10 hours
P. One train inoperable.	------------NOTE------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------------------------	
	P.1 Restore train to OPERABLE status.	6 hours
	OR	
	P.2 Be in MODE 3.	12 hours

(continued)

CP- 95-022

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Y. One, two or three Turbine Stop Valve Closure channels inoperable.	Y.1 Place channel(s) in trip. OR Y.2 Reduce THERMAL POWER to < P-9.	6 hours 10 hours
Z. Two RTS Trains inoperable	Z.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	------------------NOTES------------------ 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is $\geq$ 15% RTP. --- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours

(continued)

CP 95-022

Table 3.3.1-1 (page 2 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Source Range Neutron Flux	$2^{(d)}$	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.5 E5 cps	≤ 1.0 E5 cps
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.5 E5 cps	≤ 1.0 E5 cps
	$3^{(e)}$, $4^{(e)}$, $5^{(e)}$	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A
6. Overtemperature ΔT	1,2	4	W	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 1 (Page 3.3-21)	Refer to Note 1 (Page 3.3-21)
7. Overpower ΔT	1,2	4	W	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3-22)	Refer to Note 2 (Page 3.3-22)
8. Pressurizer Pressure						
a. Low	$1^{(f)}$	4	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1964.8 psig	≥ 1970 psig
b. High	1,2	4	W	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2390.2 psig	≤ 2385 psig

(continued)

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication.

(f) Above the P-7 (Low Power Reactor Trips Block) interlock.

CP 95-022

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
13.	SG Water Level--Low-low	1,2	3/SG	U	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.4% of narrow range span	≥ 17% of narrow range span
	Coincident with:						
	a) Vessel ΔT Equivalent to power ≤ 50% RTP	1,2	4	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
	With a time delay (Ts) if one steam generator is affected					≤ 1.01 Ts (Refer to Note 3, Page 3.3-23)	≤ Ts (Refer to Note 3, Page 3.3-23)
	or						
	A time delay (Tm) if two or more Steam Generators are affected					≤ 1.01 Tm (Refer to Note 3, Page 3.3-23)	≤ Tm (Refer to Note 3, Page 3.3-23)
	OR						
	b) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	4	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
14.	Turbine Trip						
	a. Low Fluid Oil pressure	$1^{(i)}$	3	O	SR 3.3.1.10 SR 3.3.1.14	≥ 43 psig	≥ 45 psig
	b. Turbine Stop Valve Closure	$1^{(i)}$	4	Y	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open	≥ 1% open

(continued)

(i) Above the P-9 (Power Range Neutron Flux) interlock.

CP 95-022

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low Coincident with:	1,2,3	3 per SG	M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 16.4%	≥ 17.0%
1) Vessel ΔT equivalent to power ≤ 50% RTP	1,2,3	4	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
With a time delay (Ts) if one S/G is affected					≤ 1.01 Ts (Note 1, (Page 3.3-40)	≤ Ts (Note 1, (Page 3.3-40)
or						
A time delay (Tm) if one or more S/G's are affected					≤ 1.01 Tm (Note 1, (Page 3.3-40)	≤ Tm (Note 1, (Page 3.3-40)
<u>OR</u>						
2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2,3	4	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP

(continued)

CP 95-022

BASES

APPLICABLE SAFETY ANALYSES (continued)	trip functions. RTS trip functions not specifically credited in the accident analyses have an N.A. response time requirement in Table 3.3.1-1. They are qualitatively credited in the safety analyses and the NRC staff-approved licensing basis for the plant. These RTS trip functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate function performance. These RTS trip functions may also serve as backups to RTS trip functions that were credited in the accident analysis. The safety analyses applicable to each RTS function are discussed in the bases for the Technical Specifications, B.3.3.1 (Ref. 4).
TR	OPERABILITY requirements for the RTS Instrumentation and interlocks are specified in Technical Specifications, section 3.3.1. TR 3.3.1 requires the RTS Instrumentation and interlocks of Table 3.3.1-1 of the TR to be OPERABLE with RESPONSE TIMES as shown in the table. RESPONSE TIMES must be within the specified limits for the affected instruments to be considered OPERABLE.
APPLICABILITY	Applicable MODES for the specific RTS Instrumentation and interlocks are delineated in Table 3.3.1-1 of Reference 4. The bases for Applicability of each function is included in Reference 4.
ACTIONS	<u>A.1</u> The Required Actions for inoperable instruments are found in Reference 4. With one or more RESPONSE TIMES outside the specified limits, the affected instrument(s) must be considered inoperable and the appropriate Action referenced in Table 3.3.1-1 of Reference 4 must be taken. The bases for these actions is found in Reference 4.

(continued)

CP 95-022

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides backup protection to the Power Range Neutron Flux - Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux - High Setpoint trip provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM.

(continued)

CP 95-022

BASES

ACTIONS

N.1 and N.2 (continued)

other RTS trip Functions provide core protection below the P-8 setpoint. The 6 hours allowed to restore the channel to OPERABLE status or place in trip and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 4 hour time limit is justified in Reference 7.

O.1 and O.2

Condition O applies to Turbine Trip on Low Fluid Oil Pressure. With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the tripped condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 7.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

P.1 and P.2

Condition P applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed

(continued)

CP 95-022

BASES

ACTIONS

U.1.1, U.1.2, and U.2 (continued)

place the plant in MODE 3. Six hours is a reasonable time, based on operating experience, to place the plant in MODE 3 from MODE 1 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 4 hour time limit is justified in Reference 7.

V.1 and V.2

Condition V applies to the Vessel ΔT Equivalent to Power reactor trip Function.

Failure of the vessel ΔT channel input (failure of more than one T_H RTD or failure of both T_C RTDs) affects the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

If the inoperable channel cannot be restored or the threshold power level for zero seconds time delay adjusted within the specified Completion Time, the plant must be placed in a MODE where these Functions are not required to be OPERABLE. An additional 6 hours is allowed to place the plant in MODE 3. Six hours is a reasonable time, based on operating experience, to place the plant in MODE 3 from MODE 1 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 4 hour time limit is justified in Reference 7.

(continued)

CP 95-022

BASES

ACTIONS

X.1 and X.2 (continued)

- Reactor Coolant Flow-Low (Two Loops).

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint and below the P-8 setpoint. These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition X.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 4 hour time limit is justified in Reference 7.

Y.1 and Y.2

Condition Y applies to the Turbine Trip on Stop Valve Closure. With one, two or three channels inoperable, the inoperable channels must be placed in the trip condition within 6 hours. Since all the valves must be tripped (not fully open), in order for the reactor trip signal to be generated, it is acceptable to place more than one Turbine Stop Valve Closure channel in the trip condition. With one or more channels in the trip condition, a partial reactor trip condition exists. All of the remaining Turbine Stop Valve channels are required to actuate in order to initiate a reactor trip. If a channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced to below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place an inoperable channel in the trip condition and the 4 hours allowed for reducing power are justified in Reference 7.

(continued)

CP 95-022

BASES

ACTIONS (continued)

Z.1

With two RTS trains inoperable, no automatic capability is available to shutdown the reactor, and immediate plant shutdown in accordance with the LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV. The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

The protection Functions associated with the EAGLE-21™ Process Protection System have an installed bypass capability, and may be tested in either the trip or bypass mode, as approved in Reference 7. When testing is performed in the bypass mode, the SSPS input relays are not operated, as justified in Reference 9. The input relays are checked during the CHANNEL CALIBRATION every 18 months.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the

(continued)

CP 95-022

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Low Coincident With Safety Injection and Coincident With Containment Sump Level-High (continued)

present, in addition to the SI signal and the RWST Level – Low Low signal, to transfer the suctions of the RHR pumps to the containment sump. The containment sump is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability. The containment sump level Trip Setpoint is selected to ensure enough borated water is injected to ensure the reactor remains shut down. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction. The transmitters are located inside containment and thus possibly experience adverse environmental conditions. Therefore, the Trip Setpoint reflects the inclusion of both steady state and environmental instrument uncertainties.

These Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

(continued)

CP 95-023

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

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

LCO 3.1.4 Moderator Temperature Coefficient
LCO 3.1.6 Shutdown Bank Insertion Limit
LCO 3.1.7 Control Bank Insertion Limits
LCO 3.2.1 Heat Flux Hot Channel Factor
LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3 Axial Flux Difference
LCO 3.9.1 Boron Concentration

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

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

2. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

3. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).(Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

(continued)

CP 95-024

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

CP 95-024

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs (Ref. 8), provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Steam line power operated relief valve (PORV);

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel, piping, valves, and fittings under the ASME Code, Section III, is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2735 psig.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can allow varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, ~~located, and~~ isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA) or steam generator tupe rupture (SGTR).

(continued)

CP 95-026

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump System to the condenser not be available, as discussed in the FSAR, Section 10.3 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ADV line for each of the four steam generators is provided. Each ADV line consists of one ADV and an associated block valve.

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are provided with a pressurized air supply from the auxiliary air compressors that, on a loss of pressure in the normal instrument air supply, automatically supplies backup air to operate the ADVs. The ADVs are also supplied with nitrogen to permit local operation outside the valve rooms.

A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with a DC power source and control air available. In addition, handwheels are provided for local manual operation.

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions. The capacity of the ADVs is sufficient to achieve a cooldown rate of 50°F/hr throughout the entire cooldown to RHR entry conditions with 2 ADVs in service. This permits a uniform cooldown within the capacity of the cooling water supply available in the CST.

(continued)

CP 95-027

(continued)

CP 95-027

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

PAGE	AMENDMENT	DATE
3.8-30	0	11/94
3.8-31	0	11/94
3.8-32	0	11/94
3.8-33	0	11/94
3.8-34	0	11/94
3.8-35	0	11/94
3.8-36	0	11/94
3.8-37	0	11/94
3.8-38	0	11/94
3.8-39	0	11/94
3.8-40	0	11/94
3.8-41	0	11/94
3.8-42	0	11/94
3.8-43	0	11/94
3.8-44	0	11/94
3.9-1	0	11/94
3.9-2	0	11/94
3.9-3	0	11/94
3.9-4	0	11/94
3.9-5	0	11/94
3.9-6	0	11/94
3.9-7	0	11/94
3.9-8	0	11/94
3.9-9	0	11/94
3.9-10	0	11/94
3.9-11	0	11/94
3.9-12	0	11/94
3.9-13	0	11/94
3.9-14	0	11/94
3.9-15	0	11/94
3.9-16	0	05/95
4.0-1	0	11/94
4.0-2	0	11/94
4.0-3	0	11/94
4.0-4	0	11/94
4.0-5	0	11/94
5.0-1	0	11/94
5.0-2	0	11/94
5.0-3	0	11/94
5.0-4	0	11/94
5.0-5	0	11/94
5.0-6	0	11/94
5.0-7	0	11/94
5.0-8	0	11/94

CP 95-027

TABLE OF CONTENTS (continued)

CP 95-027

LIST OF EFFECTIVE PAGES

CP 95-027

3.9 REFUELING OPERATIONS

3.9.9 Spent Fuel Pool Boron Concentration

LCO 3.9.9 Boron concentration of the spent fuel pool shall be ≥ 2000 ppm.

APPLICABILITY: During fuel movement in the flooded spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration < 2000 ppm.	A.1 Suspend fuel movement.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.9.1 Verify boron concentration in the spent fuel pool is ≥ 2000 ppm.	Prior to movement of fuel in the spent fuel pool AND 72 hours thereafter

CP 95-027

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 3.50 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

c. A nominal 10.72 inch center to center distance between fuel assemblies placed in the high density fuel storage racks.

d. With the spent fuel storage pool flooded, fuel assemblies must not be loaded in cells on the pool periphery or loaded in a configuration which would contain fuel assemblies in cells which are face adjacent.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.3 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

(continued)

CP 95-027

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 747′ - 1 1/2".

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 484 fuel assemblies.

CP 95-027

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.9.1

This SR requires that the spent fuel pool boron concentration be verified greater than or equal to 2000 ppm. This surveillance is to be performed prior to movement of fuel in the spent fuel pool and at least once ever 72 hours thereafter during the movement of fuel in the spent fuel pool.

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

REFERENCES

1. Watts Bar FSAR, Section 15, "Accident Analysis."

CP 95-028

Table 3.3.6-1 (page 1 of 1)
Containment Vent Isolation Instrumentation

	FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Manual Initiation	2	SR 3.3.6.6	NA
2.	Automatic Actuation Logic and Actuation Relays	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3.	Containment Purge Exhaust Radiation Monitors	2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	≤ 8.41E-02 μCi/cc (8.41E+04 cpm)
4.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

CP 95-030

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.3.3	Perform a functional test on a representative sample of snubbers in accordance with Table 3.7.3-4 to determine acceptance with criteria in Table 3.7.3-5.	During first refueling shutdown. AND 18 months thereafter during shutdown.
TSR 3.7.3.4	------------------NOTES---------------- 1. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. 2. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. 3. The parts replacement shall be documented and the documentation shall be retained for the duration of the unit operating license. -- Verify that the service life of hydraulic and mechanical snubbers has not been exceeded or will not be exceeded prior to the next scheduled surveillance inspection.	18 months

CP 95-030

Table 3.7.3-4 (Page 1 of 2)

Functional Testing General Notes

1. The representative sample of snubbers shall include each type and shall be tested using sample plan A for hydraulic snubbers and sample plan B for mechanical snubbers.

2. The NRC Regional Administrator shall be notified in writing of any changes to the sample plan prior to the test period.

SAMPLE PLAN A

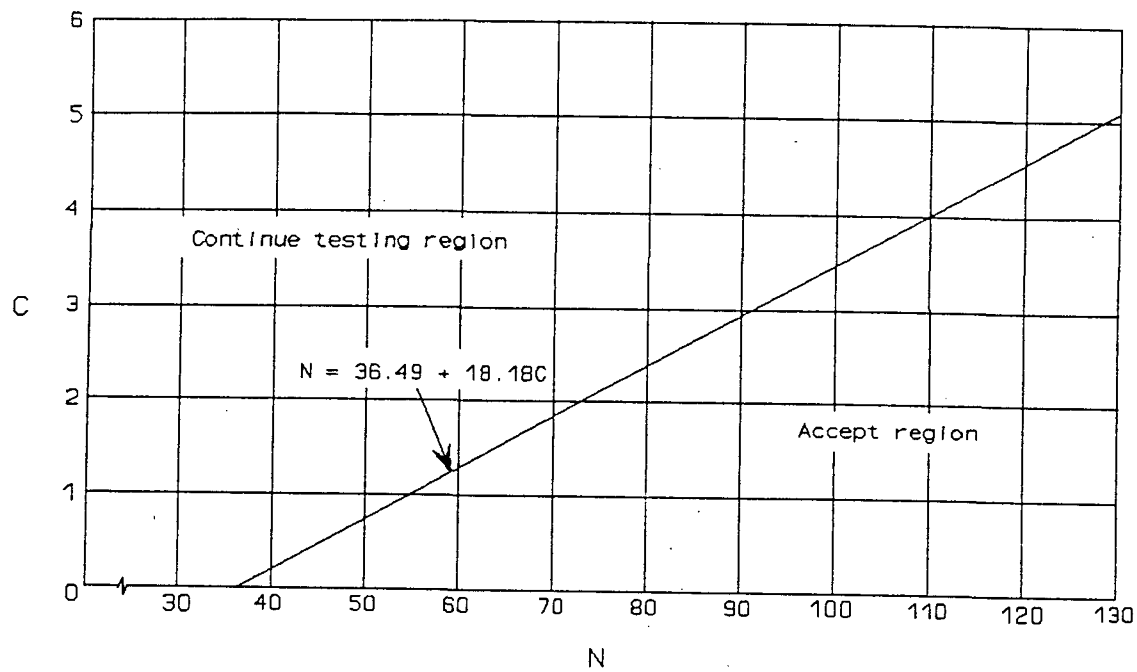
1. At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test.

2. For each snubber of a type that does not meet the functional test acceptance criteria of Table 3.7.3-5, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

SAMPLE PLAN B

1. An initial representative sample of 37 snubbers shall be functionally tested in accordance with Figure 3.7.3-1. For each snubber type which does not meet the functional test acceptance criteria of Table 3.7.3-5, another sample of at 19 snubbers shall be tested. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 36.49 + 18.18C$. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

CF 95-030



N = Total number of snubbers tested

C = Number of snubbers which do not meet functional test acceptance criteria

FIGURE 3.7.3-1

Sample Plan B for Snubber Functional Test

CP 95-032

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------NOTE------------------- Required to be performed in MODE 3. -- Verify closure time of each MSIV is $\leq$ 5.0 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program or 18 months

Table 3.3.2-1 (Page 4 of 5)

Engineered Safety Features Response Times

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air operated valves.

(4) Offsite power available - diesel generator starting and sequence loading delays not included. Response time limit includes the opening of valves to establish flowpath and bringing the pumps to full speed. The additional sequential transfer of CCP suction from the VCT to the RWST (RWST valves open, then the VCT valves close) is included.

(5) Diesel generator starting and sequence loading delays included. Response time limit includes the opening of valves to establish flow path and bringing the pumps up to full speed. The additional sequential transfer of suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(6) The following equipment are exceptions to the response time shown in the table and will have the following response times for the initiating signals and functions:

A. Fire Protection CIVs	22(2)/32(1)
B. Ice Condenser CIVs	32
C. Excess Letdown Hx Supply CIV	68(2)/78(1)
D. EGTS Fans	20(2)/30(1)
E. Required for EGTS OPERABILITY	
1. Fire Protection Secondary CIVs	20(2)/30(1)
2. Secondary Containment Purge Isolation Valves	12.7(2)/22.7(1)
CP 95-034 → F. Steam Generator Blowdown CIVs	17(2)/27(1)

(7) On 2/3 any steam generator and Trip Time Delay = 0 seconds.

(8) On 2/3 in 2/4 steam generators and Trip Time Delay = 0 seconds.

(9) The response time is measured from the time the 6.9 kV shutdown boards voltage exceeds the Setpoint until the time full voltage is returned for the loss of voltage sensors; or from the time the degraded voltage timers generate a signal to trip the feeder breakers and shed loads until the time full voltage is returned for the degraded voltage sensors.

(10) The Response Time for motor-driven AFW pumps includes the diesel generator starting and sequence loading delays. The Response Time for (steam) turbine driven AFW pumps does not include diesel generator starting and sequence loading delays.

(continued)

CP 95-035

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.3.1	Verify that the upstream manual RCSV isolation valve is locked in the opened position.	18 months
TSR 3.4.3.2	Operate each remotely controlled valve through at least one complete cycle of full travel from the control room.	In accordance with the Inservice Testing Program
TSR 3.4.3.3	Verify flow through the RCSV paths during venting.	18 months

CP 95-035

BASES

ACTIONS

B.1 (continued)

and correcting problems which could be associated with an inoperable path.

C.1 AND C.2

If the Required Action and associated Completion Time of Condition A or B are not met, the plant must be placed in a condition in which the TR does not apply. This is accomplished by placing the unit in MODE 3 within 6 hours and MODE 4 in an additional 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.3.1

Every 92 days it is necessary to verify the function of the block valve to isolate a failed open RCS vent valve. Cycling the block valve closed and open demonstrates its capability to perform this function. These valves are required to seat against full operational pressure with the down stream side at the pressure of the Pressurizer Relief Tank. The valves are solenoid-to-open and spring-to-close valves.

TSR 3.4.3.1, TSR 3.4.3.2 and TSR 3.4.3.3

Every 18 months it is necessary to verify that each of the two vent paths are OPERABLE. This verification consists of checking the upstream isolation valve and ensuring that the valve is locked in the open position. Further, the two control valves are operated from the control room, in accordance with the Inservice Testing Program through one complete cycle of full travel. Lastly, the test includes a verification of flow through the two vent paths.

REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements".
2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

CP 95-037

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	-------------------NOTE------------------- Required to be performed once per 184 days within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

CP 95-037

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed once per 184 days within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," 1973.
2. Watts Bar FSAR, Section 15.4, "Condition IV - Limiting Faults."

CP 95-040

Table 3.8.3-1 (Page 1 of 5)

Motor-Operated Valves Thermal Overload Devices Which Are Bypassed Under Accident Conditions

VALVE NO.	FUNCTION
1-FCV-62-63	Isolation for Seal Water Filter
1-FCV-62-90	ECCS Operation
1-FCV-62-91	ECCS Operation
1-FCV-62-61	Cont. Isolation
1-LCV-62-132	ECCS Operation
1-LCV-62-133	ECCS Operation
1-LCV-62-135	ECCS Operation
1-LCV-62-136	ECCS Operation
1-FCV-74-1	Open for Normal Plant Cooldown
1-FCV-74-2	Open for Normal Plant Cooldown
1-FCV-74-3	ECCS Operation
1-FCV-74-21	ECCS Operation
1-FCV-74-12	RHR Pump, Mini-flow Protects Pump
1-FCV-74-24	RHR Pump, Mini-flow Protects Pump
1-FCV-74-33	ECCS Operation
1-FCV-74-35	ECCS Operation
1-FCV-63-7	ECCS Operation
1-FCV-63-6	ECCS Operation
1-FCV-63-156	ECCS Flow Path
1-FCV-63-157	ECCS Flow Path
1-FCV-63-25	BIT Injection
1-FCV-63-26	BIT Injection
1-FCV-63-1	ECCS Operation
1-FCV-63-72	ECCS Flow Path from Cont. Sump
1-FCV-63-73	ECCS Flow Path from Cont. Sump
1-FCV-63-8	ECCS Flow Path
1-FCV-63-11	ECCS Flow Path
1-FCV-63-93	ECCS Cooldown Flow Path
1-FCV-63-94	ECCS Cooldown Flow Path

(continued)

CP 95-040

Table 3.8.3-1 (Page 2 of 5)

Motor-Operated Valves Thermal Overload
Devices Which Are Bypassed Under
Accident Conditions

(continued)

VALVE NO.	FUNCTION
1-FCV-63-172	ECCS Flow Path
1-FCV-63-5	ECCS Flow Path
1-FCV-63-47	Train Isolation
1-FCV-63-48	Train Isolation
1-FCV-63-4	SI Pump Mini-flow
1-FCV-63-175	SI Pump Mini-flow
1-FCV-63-3	SI Pump Mini-flow
1-FCV-63-152	ECCS Recirc
1-FCV-63-153	ECCS Recirc
1-FCV-3-33	Quick Closing Isolation
1-FCV-3-47	Quick Closing Isolation
1-FCV-3-87	Quick Closing Isolation
1-FCV-3-100	Quick Closing Isolation
1-FCV-1-15	Steam Supply to Aux FWP Turbine
1-FCV-1-16	Steam Supply to Aux FWP Turbine
1-FCV-3-179A	ERCW System Supply to Pump
1-FCV-3-179B	ERCW System Supply to Pump
1-FCV-3-136A	ERCW System Supply to Pump
1-FCV-3-136B	ERCW System Supply to Pump
1-FCV-3-116A	ERCW System Supply to Pump
1-FCV-3-116B	ERCW System Supply to Pump
1-FCV-3-126A	ERCW System Supply to Pump
1-FCV-3-126B	ERCW System Supply to Pump
1-FCV-70-133	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-143	Isolation for Excess Letdown Ht Xchngr
1-FCV-70-92	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-90	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-87	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-89	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-140	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-134	Isolation for RCP Oil Coolers & Therm B

(continued)

CP 95-040

Table 3.8.3-1 (Page 3 of 5)

Motor-Operated Valves Thermal Overload Devices Which Are Bypassed Under Accident Conditions

(continued)

VALVE NO.	FUNCTION
1-FCV-67-123	CS Heat Exchanger Supply
1-FCV-67-125	CS Heat Exchanger Supply
1-FCV-67-124	CS Heat Exchanger Discharge
1-FCV-67-126	CS Heat Exchanger Discharge
1-FCV-67-146	CCWS Heat Exchanger Throttling
1-FCV-67-83	Containment Isolation Lower
1-FCV-67-88	Containment Isolation Lower
1-FCV-67-87	Containment Isolation Lower
1-FCV-1-51	AFPT Trip and Throttle Valve
1-FCV-67-95	Containment Isolation Lower
1-FCV-67-96	Containment Isolation Lower
1-FCV-67-91	Containment Isolation Lower
1-FCV-67-103	Containment Isolation Lower
1-FCV-67-104	Containment Isolation Lower
1-FCV-67-99	Containment Isolation Lower
1-FCV-67-111	Containment Isolation Lower
1-FCV-67-112	Containment Isolation Lower
1-FCV-67-107	Containment Isolation Lower
1-FCV-67-130	Containment Isolation Upper
1-FCV-67-131	Containment Isolation Upper
1-FCV-67-295	Containment Isolation Upper
1-FCV-67-134	Containment Isolation Upper
1-FCV-67-296	Containment Isolation Upper
1-FCV-67-133	Containment Isolation Upper
1-FCV-67-139	Containment Isolation Upper
1-FCV-67-297	Containment Isolation Upper
1-FCV-67-138	Containment Isolation Upper
1-FCV-67-142	Containment Isolation Upper
1-FCV-67-298	Containment Isolation Upper

(continued)

CP 95-040

Table 3.8.3-1 (Page 4 of 5)

Motor-Operated Valves Thermal Overload
Devices Which Are Bypassed Under
Accident Conditions

(continued)

VALVE NO.	FUNCTION
1-FCV-67-141	Containment Isolation Upper
1-FCV-72-21	Cont. Spray Pump Suction
1-FCV-72-22	Cont. Spray Pump Suction
1-FCV-72-2	Cont. Spray Isolation
1-FCV-72-39	Cont. Spray Isolation
1-FCV-72-40	RHR Cont. Spray Isolation
1-FCV-72-41	RHR Cont. Spray Isolation
1-FCV-72-44	Cont. Sump to Hdr A - Cont. Spray
1-FCV-72-45	Cont. Sump to Hdr B - Cont. Spray
1-FCV-26-240	Containment Isolation
1-FCV-26-241	Annulus Isolation
1-FCV-26-242	Annulus Isolation
1-FCV-26-243	RCP Cont. Spray Isolation
1-FCV-26-244	Annulus Isolation
1-FCV-26-245	Annulus Isolation
1-FCV-68-332	RCS PRZR Rel.
1-FCV-68-333	RCS PRZR Rel.
1-FCV-70-153	RHR Ht Ex B-B Outlet
1-FCV-70-156	RHR Ht Ex A-A Outlet
1-FCV-70-207	Cont. Demin. Waste Evap. Bldg Supply
1-FCV-67-9A	ERCW Strainer Backwash
2-FCV-67-9A	ERCW Strainer Backwash
1-FCV-67-9B	ERCW Strainer Flush
2-FCV-67-9B	ERCW Strainer Flush
1-FCV-67-10A	ERCW Strainer Backwash
2-FCV-67-10A	ERCW Strainer Backwash
1-FCV-67-10B	ERCW Strainer Flush
2-FCV-67-10B	ERCW Strainer Flush

(continued)

CP 95-040

Table 3.8.3-1 (Page 5 of 5)

Motor-Operated Valves Thermal Overload
Devices Which Are Bypassed Under
Accident Conditions

(continued)

VALVE NO.	FUNCTION
1-FCV-67-89	Containment Isolation
1-FCV-67-97	Containment Isolation
1-FCV-67-105	Lower Containment Isolation
1-FCV-67-113	Lower Containment Isolation
1-FCV-67-143	CCS Heat Exchanger Discharge
0-FCV-67-144	CCS Heat Exchanger Bypass
0-FCV-67-152	CCS Heat Exchanger Discharge
0-FCV-67-205	Nonessential Equipment Isolation
0-FCV-67-208	Station Service/Contr. Air Supply
1-FCV-70-183	Sample Ht Ex Header Outlet
1-FCV-70-100	RCP Oil Cooler Supply Cont. Isolation
0-FCV-70-197	SFPCS Ht Ex Supply Header
1-FCV-70-215	Sample Ht Ex Header Inlet
1-FCV-74-8	RHR Isolation Bypass
1-FCV-74-9	RHR Isolation Bypass

CP 95-041

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two ABGTS trains inoperable during movement of irradiated fuel assemblies in the fuel handling area.	D.1 Suspend movement of irradiated fuel assemblies in the fuel handling area.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABGTS train for $\geq$ 10 continuous hours with the heaters operating.	31 days
SR 3.7.12.2	Perform required ABGTS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.12.4	Verify one ABGTS train can maintain a pressure between -0.25 and -0.5 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate $\geq$ 9300 and $\leq$ 9900 cfm.	18 months on a STAGGERED TEST BASIS

CP 95-041

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 8.

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 and -0.5 inches water gauge with respect to atmospheric pressure at a nominal flow rate ≥ 9300 and ≤ 9900 cfm. Periodic testing of ABGTS shall be performed once with one AB general supply fan running and one train of its discharge ABSCE isolation dampers open and once with one purge air supply fan running and one of its suction-side ABSCE isolation dampers open. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 9).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 8.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, "Engineered Safety Feature (ESF) Filter Systems."
2. Watts Bar FSAR, Section 9.4.2, "Fuel Handling Area Ventilation System."
3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."

(continued)

CP 95-042

TR 3.7 PLANT SYSTEMS

TR 3.7.2 Flood Protection Plan

TR 3.7.2 The flood protection plan shall be ready for implementation to maintain the plant in a safe condition.

APPLICABILITY: When one or more of the following conditions exist:

a. Flood-producing rainfall conditions in the east Tennessee watershed, or

b. An early warning or alert that a critical combination of flood and/or high headwater levels may or have developed, or

c. An early warning or alert involving Fontana Dam, or

d. Recognizable seismic activity in the east Tennessee region.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Stage I flood warning issued.	A.1 Be in at least MODE 3.	6 hours
	<u>AND</u>	
	A.2 Initiate and complete the Stage I flood protection plan.	10 hours
	<u>AND</u>	
	A.3 Establish a SDM $\geq$ 5% $\Delta k/k$ and $T_{avg} \leq$ 350°F.	10 hours
	<u>AND</u>	
		(continued)

95-042

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4.1 Verify that communications between the TVA Division of Water Resources and the Watts Bar Nuclear Plant have been established. OR A.4.2 Initiate and complete the Stage II flood protection procedure.	10 hours 27 hours
B. Stage II flood warning issued.	B.1 Initiate and complete the Stage II flood protection plan.	17 hours OR Prior to predicted flooding of the site.
C. Seismic event occurs after a critical combination of flood and/or headwater alerts are issued.	C.1 Verify and maintain communications between TVA Power Control Center and the Watts Bar Nuclear Plant. OR	6 hours

(continued)

CP 95-042

B 3.7 PLANT SYSTEMS

B 3.7.2 Flood Protection Plan

BASES

BACKGROUND

Nuclear power plants are designed to prevent the loss of capability for cold shutdown and maintenance thereof resulting from the most severe flood conditions that can reasonably be predicted to occur at the site as a result of severe hydrometeorological conditions, seismic activity, or both (Ref. 1). Assurance that safety-related facilities are capable of surviving all possible flood conditions is provided by the flood protection plan.

The elevations of plant features which could be affected by the submergence during floods vary from 714.5 ft Mean Sea Level (MSL) (access to electrical conduits) to 740.1 ft MSL (including wave runup). Plant grade is elevation 728 ft MSL which can be exceeded by rainfall floods and by seismic-caused dam failure floods. One kind of warning plan is needed to assure plant safety from rainfall floods, and another kind of warning plan is needed for seismic-caused dam failure floods.

The warning plan is divided into two stages. This two-stage scheme is designed to prevent excessive economic loss in case a potential flood does not fully develop. Stage I, which is a minimum of 10 hours, allows preparation steps, causing some damage to be sustained, but will postpone major economic damage. Stage II, which is a minimum of 17 hours, is a warning that assumes a forthcoming flood above grade. The time limits on the stages are given so that unnecessary economic penalty can be avoided while adequate time is allowed for preparing for operation in the flood mode.

Stage I procedures consist of a controlled reactor shutdown and other easily revokable steps, such as moving flood supplies above the maximum possible flood elevation and making temporary connections and load adjustments on the onsite power supply. After unit shutdown, the Reactor Coolant System will be cooled and the pressure will be reduced to less than 350 psig. Stage II procedures are the least easily revokable and more damaging steps necessary to have the plant in the flood mode when the flood exceeds plant grade. Heat removal from the steam generators will be accomplished by adding river water

(continued)

CP 95-042

ACTIONS (continued)

with the prospect of reaching elevation of 727 ft MSL, 1 foot below plant grade, are early enough to assure adequate warning time for safe plant shutdown.

B.1

If the Stage II flood warning has been issued, the Stage II flood protection plan must be initiated and completed within 17 hours or prior to the predicted flooding of the site. The Completion Time of 17 hours corresponds to the remaining hours of the 27 hour preflood preparation time after the Stage I flood warning consisting of 10 hours has expired, and is an adequate time period to complete Stage II preparations.

C.1, C.2.1, C.2.2.1, and C.2.2.2

If a seismic event occurs after a critical combination of flood and/or headwater alerts is issued, within 6 hours communications between TVA Power Control Center and the Watts Bar Nuclear Plant must be verified and maintained.The TVA Power Control Center is able to detect unexplained electrical interruptions at dams (not including Fontana Dam), or loss of contact with the dams involved in the issued alert. If an unexplained interruption occurs, the Watts Bar Plant Manager will be notified and efforts will be made by the TVA Power Control Center to determine whether dam failure has occurred. The 6-hour Completion Time is an adequate time period to complete the requirements of Required Action C.1.

If Required Action C.1 and the associated Completion Time is not met, the Stage I flood protection plan must be initiated and completed within the 16 hours. The Completion Time for this Required Action is 16 hours which is adequate time for preparing for operation in the flood mode.

Also, communications between the TVA Power Control Center and Watts Bar Nuclear Plant must be established prior to the completion of the Stage I flood protection plan. If communications cannot be established, the Stage II flood protection plan must be initiated and completed within 17 additional hours (33 hours total). The Completion Time of 33 hours corresponds the TVA Division of Water Resources preflood preparation time and is an adequate time period to complete shutdown.

(continued)

CP 95-043

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling System (CCS)

BASES

BACKGROUND

The CCS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCS also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCS serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Raw Cooling Water (ERCW) System, and thus to the environment.

The CCS is arranged as two independent, full-capacity cooling trains. Separate Train A equipment is provided in each unit, whereas Train B is shared by both units. Train A in unit 1 is served by CCS Hx A and CCS pump 1A-A. Pump 1B-B, which is actually Train B equipment, is also normally aligned to the Train A header in unit 1. However, pump 1B-B can be realigned to Train B on loss of Train A.

Similarly, Train A in unit 2 is served by CCS Hx B and CCS pump 2A-A with support from pump 2B-B.

Train B in both units is served by CCS Hx C. Normally, only CCS pump C-S is aligned to the Train B headers since few nonessential, normally-operating loads are assigned to Train B. However, pumps 1B-B or 2B-B can be realigned to the Train B headers on a loss of the C-S pump. CCS pump 2B-B is aligned with CCS pump C-S and Train 1B during Unit 1 only operation.

Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential components will be manually isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the CCS is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post accident cooldown and shutdown.

(continued)

CP 94-019

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
13. SG Water Level--Low-low Coincident with:	1,2	3/SG	U	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.4% of narrow range span	≥ 17% of narrow range span
a) Vessel ΔT Equivalent to power ≤ 50% RTP	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
With a time delay (Ts) if one steam generator is affected					≤ 1.01 Ts (Refer to Note 3, Page 3.3-23)	≤ Ts (Refer to Note 3, Page 3.3-23)
or						
A time delay (Tm) if two or more Steam Generators are affected					≤ 1.01 Tm (Refer to Note 3, Page 3.3-23)	≤ Tm (Refer to Note 3, Page 3.3-23)
OR						
b) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
14. Turbine Trip						
a. Low Fluid Oil pressure	$1^{(i)}$	3	O	SR 3.3.1.10 SR 3.3.1.14	≥ 43 psig	≥ 45 psig
b. Turbine Stop Valve Closure	$1^{(i)}$	4	Y	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open	≥ 1% open

(continued)

(i) Above the P-9 (Power Range Neutron Flux) interlock.

CP 94-019

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low	1,2,3	3 per SG	M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 16.4%	≥ 17.0%
Coincident with:						
1) Vessel ΔT equivalent to power ≤ 50% RTP	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
With a time delay (Ts) if one S/G is affected					≤ 1.01 Ts (Note 1, (Page 3.3-40)	≤ Ts (Note 1, (Page 3.3-40)
or						
A time delay (Tm) if one or more S/G's are affected					≤ 1.01 Tm (Note 1, (Page 3.3-40)	≤ Tm (Note 1, (Page 3.3-40)
<u>OR</u>						
2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP

(continued)

CP 94-019

Table 3.3.2-1 (page 6 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
7. Automatic Switchover to Containment Sump (continued)						
b. Refueling Water Storage Tank (RWST) Level – Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 155.6 inches from Tank Base	≥ 158 inches from Tank Base
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
and						
Coincident with Containment Sump Level – High	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 37.7 in. above el. 702.8 ft	≥ 38.2 in. above el. 702.8 ft
8. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	F	SR 3.3.2.11	NA	NA
b. Pressurizer Pressure, P-11						
(1) Unblock (Auto Reset of SI Block)	1,2,3	3	L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9	≤ 1975.2 psig	≤ 1970 psig
(2) Enable Manual Block of SI	1,2,3	3	L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9	≥1956.8 psig	≥ 1962 psig

CP 94-019

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the Watts Bar setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of sensor/transmitter drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. For channels with a trip time delay (TTD), this test shall include verification that the TTD coefficients are adjusted correctly.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, 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 power range neutron detectors consists of a normalization of the detectors based on a power calorimetric performed above 15% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and

(continued)

CP 94-019

SURVEILLANCE REQUIREMENTS

SR 3.3.2.9 (continued)

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of sensor/transmitter drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable. For channels with a trip time delay (TTD), this test shall include verification that the TTD coefficients are adjusted correctly.

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in Technical Requirements Manual, Section 3.3.2 (Ref. 8). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of sequential tests such that the entire response time is measured.

ESF RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel.

(continued)

CP 95-045

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMIT FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ $\frac{1}{4}$ inch above maximum level indication mark[(a)]	> Minimum level indication mark, and ≤ $\frac{1}{4}$ inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[(b)(c)]	≥ 1.200	≥ 1.195 AND Average of all connected cells > 1.205	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge for vital batteries and < 1.0 amp for DG batteries.

(c) A battery charging current of < 2 amps when on float charge for vital batteries and < 1.0 amp for DG batteries is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 31 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 31 day allowance.

CP 95-045

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

manufacturer recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

The footnotes to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote b to Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge for vital batteries and < 1.0 amps for DG batteries. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 2). Footnote c to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 31 days following a battery recharge. Within 31 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 31 days.

REFERENCES

1. Watts Bar FSAR, Section 15, "Accident Analysis," and Section 6, "Engineered Safety Features."

2. IEEE-450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

EDITORIAL CHANGES

Table 3.3.2-1 (page 5 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
c. Safety Injection	Refer to Function 1 (safety Injection) for all initiation functions and requirements					
d. Loss of Offsite Power	1,2,3	4 per bus	F	Refer to Function 4 of Table 3.3.5-1 for SRs and Allowable Values		
e. Trip of all Main Feedwater Pumps	1,2	1 per pump	J	SR 3.3.2.8 SR 3.3.2.9 SR 3.3.2.10	≥ 48 psig	≥ 50 psig
f. Motor-Driven Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure —Low	1,2,3	3	F	SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	A) ≥ 0.5 psig B) ≥ 1.33 psig	A) ≥ 1.2 psig B) ≥ 2.0 psig
g. Turbine-driven AFW Pump Suction Train A and B Transfer on Suction Pressure--Low	1,2,3	2/train, 2 trains	F	SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	A) ≥ 11.6 psig B) ≥ 12.2 psig	A) ≥ 12.8 psig B) ≥ 13.5 psig
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA

(continued)

(g) Setpoint verification not required.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1	Increase RCS cold leg temperature to > 350°F.	12 hours
	OR		
	D.2	Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
E. One required RCS relief valve inoperable in MODE 4.	E.1	Restore required RCS relief valve to OPERABLE status.	7 days
F. One required RCS relief valve inoperable in MODE 5 or 6.	F.1	Restore required RCS relief valve to OPERABLE status.	24 hours
G. Two required RCS relief valves inoperable. OR Required Action and associated Completion Time of Condition A, B, D, E, or F not met. OR COMS inoperable for any reason other than Condition A, B, C, D, E, or F.	G.1	Depressurize RCS and establish RCS vent.	8 hours

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.4 (continued)

The Frequency of 92 days is justified in Reference 7, except for Function 7. The Frequency for Function 7 is justified in Reference 10.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 92 days. The Frequency is adequate, based on industry operating experience, considering instrument reliability and operating history data.

This SR is modified by a Note, which states that performance of this test is not required for those relays tested by SR 3.3.2.7.

SR 3.3.2.6

SR 3.3.2.6 is the performance of a TADOT every 92 days. This test is a check of the Loss of Offsite Power (Function 6.d), AFW Pump Suction Transfer on Suction Pressure – Low for motor driven and turbine driven pumps (Functions 6.f and 6.g respectively), and Turbine Trip and Feedwater Isolation - Main Steam Valve Vault Rooms Water Level - High (Function 5.d).

The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. The Frequency is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

(continued)

BASES

ACTIONS (continued)

E.1

Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a waste gas decay tank rupture.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREVS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

(continued)

BASES

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 17, "Electric Power System."

2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," U.S. Nuclear Regulatory Commission, March 10, 1971.

3. IEEE-308-1971, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.

4. Watts Bar FSAR, Section 8.3.2, "DC Power System."

5. IEEE-485-1983, "Recommended Practices for Sizing Large Lead Storage Batteries for Generating Stations and Substations," Institute of Electrical and Electronic Engineers.

6. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," February 1977, U.S. Nuclear Regulatory Commission.

7. Watts Bar FSAR, Section 15, "Accident Analysis" and Section 6 "Engineered Safety Features."

8. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

9. IEEE-450-1980, "IEEE Recommended Practice for Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," Institute of Electrical and Electronic Engineers.

10. TVA Calculation WBN EEB-MS-TI11-0003, "125 VDC Vital Battery and Charger Evaluation."

11. Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," U.S. Nuclear Regulatory Commission, February 1978.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, vital DC, and AC vital bus electrical power distribution subsystems are functioning properly, with all the required buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. Watts Bar FSAR, Section 8.0, "Electric Power," Section 15, "Accident Analysis," and Section 6, "Engineered Safety Features."

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.

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 moved to 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) System pumps.

The pumping action of the 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

(continued)

BASES

BACKGROUND (continued)

3. Fuel Burnup - boron concentration must be decreased to compensate for fuel burnup and the buildup of fission products in the fuel.

4. Cold Shutdown - boron concentration must be increased to the cold shutdown concentration.

Boric acid is stored in three boric acid tanks. Two boric acid transfer pumps are provided for each unit with one pump normally aligned with one boric acid tank and continuously running at low speed to provide recirculation for the boric acid system and the boric acid tank. On a demand signal by the reactor makeup control system, the boric acid transfer pumps are shifted to high speed and the pump aligned to the makeup system delivers boric acid to the suction header of the charging pumps (Ref. 1).

APPLICABLE SAFETY ANALYSES

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. In the case of a malfunction of the CVCS, which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 2). OPERABILITY of the charging pumps, the RWST, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.

TR

TR 3.1.1 requires at least one boron injection flow path to be OPERABLE and capable of being powered from an OPERABLE emergency power source during MODES 4, 5, and 6 in order to provide a path to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations. This requirement may be achieved by meeting one of the following two conditions:

a. A flow path from an OPERABLE boric acid storage tank, through the boric acid transfer pump, through a charging pump to the RCS, or

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.2 Communications

BASES

BACKGROUND	During CORE ALTERATIONS communication ability must be retained between the control room and personnel on the refueling station. This is needed to allow the refueling personnel to be informed of any significant changes in the unit status or core reactivity conditions.
APPLICABLE SAFETY ANALYSES	This requirement helps assure direct communications between the control room and refueling personnel during refueling, which would help to preclude inadvertent criticality. It also ensures that the refueling personnel are able to inform the control room if there are any problems or accidents during the refueling process. Refueling operations are not addressed in PRA studies and would not be important in accident sequences that are commonly found to dominate risk (Ref. 1).
TR	TR 3.9.2 requires that direct communications be maintained between the control room and personnel at the refueling station. This ensures that information can be exchanged between the two groups if any unplanned events occur or if any significant changes occur in the unit status or core reactivity conditions.
APPLICABILITY	TR 3.9.2 is only applicable during CORE ALTERATIONS (MODE 6). In all other MODES refueling procedures do not take place and are therefore not applicable.

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;
2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;
3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;
4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized in advance by the Plant Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The Operations Manager shall hold or have held an SRO license on a similar unit.

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Operations Supervisor (SOS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on shift (Generic Letter 86-04 dated 02/13/86).

TR 3.4 REACTOR COOLANT SYSTEM (RCS)

TR 3.4.3 Reactor Coolant System Vents

TR 3.4.3 Two Reactor Coolant System Vent (RCSV) paths shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCSV path inoperable.	A.1 Initiate action to maintain the affected RCSV path closed with power removed from the valve actuators. AND A.2 Restore the inoperable path to OPERABLE status.	Immediately 30 days
B. Two RCSV paths inoperable.	B.1 Restore one RCSV path to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

BASES

ACTIONS

B.1 (continued)

and correcting problems which could be associated with an inoperable path.

C.1 AND C.2

If the Required Action and associated Completion Time of Condition A or B are not met, the plant must be placed in a condition in which the TR does not apply. This is accomplished by placing the unit in MODE 3 within 6 hours and MODE 5 in an additional 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.3.1

Every 92 days it is necessary to verify the function of the block valve to isolate a failed open RCS vent valve. Cycling the block valve closed and open demonstrates its capability to perform this function. These valves are required to seat against full operational pressure with the down stream side at the pressure of the Pressurizer Relief Tank. The valves are solenoid-to-open and spring-to-close valves.

TSR 3.4.3.1, TSR 3.4.3.2 and TSR 3.4.3.3

Every 18 months it is necessary to verify that each of the two vent paths are OPERABLE. This verification consists of checking the upstream isolation valve and ensuring that the valve is locked in the open position. Further, the two control valves are operated from the control room, in accordance with the Inservice Testing Program through one complete cycle of full travel. Lastly, the test includes a verification of flow through the two vent paths.

REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements".
2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

ENCLOSURE 4

REVISED PAGES FOR

INCLUSION IN FINAL DRAFT

(continued)

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
i	0	11/94
ii	0	11/94
iii	0	05/95
iv	0	11/94
v	0	11/94
vi	0	11/94
vii	0	11/94
viii	0	05/95
ix	0	05/95
x	0	05/95
xi	0	05/95
xii	0	05/95
xiii	0	05/95
xiv	0	05/95
xv	0	05/95
xvi	0	05/95

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1.3-12	0	11/94
1.3-13	0	11/94
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1.4-2	0	11/94

TECHNICAL SPECIFICATIONS

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3.2-11	0	11/94
3.2-12	0	11/94

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

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3.3-44	0	11/94

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

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3.4-22	0	11/94
3.4-23	0	11/94
3.4-24	0	11/94

LIST OF EFFECTIVE PAGES

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

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3.8-28	0	11/94
3.8-29	0	11/94

LIST OF EFFECTIVE PAGES

TECHNICAL SPECIFICATIONS

LIST OF EFFECTIVE PAGES

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5.0-36	0	11/94
5.0-37	0	11/94
5.0-38	0	11/94
5.0-39	0	11/94

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.7.2.18, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.6 (continued)	When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.
LCO 3.0.7	Test Exception LCOs 3.1.9, 3.1.10, and 3.4.17 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 SR APPLICABILITY

SR 3.0.3 (continued)	When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM) – T_{avg} > 200°F

LCO 3.1.1 SDM shall be ≥ 1.6% $\Delta k/k$.

APPLICABILITY: MODE 2 with k_{eff} < 1.0,
MODES 3 and 4.

-----NOTE-----
While this LCO is not met, entry into MODE 4 from MODE 3 is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is ≥ 1.6% $\Delta k/k$.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 SHUTDOWN MARGIN (SDM) – $T_{avg} \le 200°F$

LCO 3.1.2 The SDM shall be ≥ 1.0% Δk/k.

APPLICABILITY: MODE 5.

-----------------------------NOTE--------------------------
While this LCO is not met, entry into MODE 5 from MODE 4 is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.1 Verify SDM is ≥ 1.0% Δk/k.	24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel or train inoperable.	C.1 Restore channel or train to OPERABLE status. OR C.2 Open RTBs.	48 hours 49 hours
D. One Power Range Neutron Flux – High channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. ---------------------------- D.1.1 Place channel in trip. AND D.1.2 Reduce THERMAL POWER to ≤ 75% RTP. OR D.2.1 Place channel in trip. AND ------------NOTE------------ Only required to be performed when the Power Range Neutron Flux input to QPTR is inoperable. ---------------------------- D.2.2 Perform SR 3.2.4.2. OR D.3 Be in MODE 3.	 6 hours 12 hours 6 hours Once per 12 hours 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
O. One Low Fluid Oil Pressure Turbine Trip channel inoperable.	------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------------------------	
	O.1 Place channel in trip.	6 hours
	OR	
	O.2 Reduce THERMAL POWER to < P-9.	10 hours
P. One train inoperable.	------------NOTE------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------------------------	
	P.1 Restore train to OPERABLE status.	6 hours
	OR	
	P.2 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
Y. One, two or three Turbine Stop Valve Closure channels inoperable.	Y.1	Place channel(s) in trip.	6 hours
	<u>OR</u>		
	Y.2	Reduce THERMAL POWER to < P-9.	10 hours
Z. Two RTS Trains inoperable	Z.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	-------------------NOTES------------------- 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. ---	
	Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.7	------------------NOTE------------------- Not required to be performed for Source Range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. -- Perform COT.	92 days (31 days for Function 5)
SR 3.3.1.8	------------------NOTE------------------- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -- Perform COT.	------NOTE----- Only required when not performed within previous 92 days (31 days for Functions 4 and 5) -------------- Prior to reactor startup **AND** Four hours after reducing power below P-10 for power and intermediate range instrumentation **AND**

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8 (continued)		Four hours after reducing power below P-6 for source range instrumentation AND Every 92 days thereafter (31 days for Functions 4 and 5)
SR 3.3.1.9	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	92 days
SR 3.3.1.10	-------------------NOTE------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.13	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	18 months
SR 3.3.1.14	-------------------NOTE------------------- Verification of setpoint is not required. -- Perform TADOT.	-----NOTE------ Only required when not performed within previous 31 days --------------- Prior to reactor startup
SR 3.3.1.15	-------------------NOTE------------------- Neutron detectors are excluded from response time testing. -- Verify RTS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.1-1 (page 2 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Source Range Neutron Flux	2[(d)]	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.5 E5 cps	≤ 1.0 E5 cps
	3[(a)], 4[(a)], 5[(a)]	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.5 E5 cps	≤ 1.0 E5 cps
	3[(e)], 4[(e)], 5[(e)]	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A
6. Overtemperature ΔT	1,2	4	W	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 1 (Page 3.3-21)	Refer to Note 1 (Page 3.3-21)
7. Overpower ΔT	1,2	4	W	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3-22)	Refer to Note 2 (Page 3.3-22)
8. Pressurizer Pressure						
a. Low	1[(f)]	4	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1964.8 psig	≥ 1970 psig
b. High	1,2	4	W	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2390.2 psig	≤ 2385 psig

(continued)

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication.

(f) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
13. SG Water Level--Low-low Coincident with:	1,2	3/SG	U	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.4% of narrow range span	≥ 17% of narrow range span
a) Vessel ΔT Equivalent to power ≤ 50% RTP	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
With a time delay (Ts) if one steam generator is affected					≤ 1.01 Ts (Refer to Note 3, Page 3.3-23)	≤ Ts (Refer to Note 3, Page 3.3-23)
or						
A time delay (Tm) if two or more Steam Generators are affected					≤ 1.01 Tm (Refer to Note 3, Page 3.3-23)	≤ Tm (Refer to Note 3, Page 3.3-23)
<u>OR</u>						
b) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
14. Turbine Trip						
a. Low Fluid Oil pressure	1(i)	3	O	SR 3.3.1.10 SR 3.3.1.14	≥ 43 psig	≥ 45 psig
b. Turbine Stop Valve Closure	1(i)	4	Y	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open	≥ 1% open

(continued)

(i) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 5 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	P	SR 3.3.1.13	NA	NA
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2(d)	2	R	SR 3.3.1.11 SR 3.3.1.12	≥ 1.08E-04% RTP	≥ 1.66E-04% RTP
b. Low Power Reactor Trips Block, P-7	1	1 per train	S	SR 3.3.1.11 SR 3.3.1.12	NA	NA
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 50.4% RTP	≤ 48% RTP
d. Power Range Neutron Flux, P-9	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 52.4% RTP	≤ 50% RTP
e. Power Range Neutron Flux, P-10	1,2	4	R	SR 3.3.1.11 SR 3.3.1.12	≥ 7.6% RTP and ≤ 12.4% RTP	≥ 10% RTP
f. Turbine Impulse Pressure, P-13	1	2	S	SR 3.3.1.10 SR 3.3.1.12	≤ 12.4% full-power pressure	≤ 10% full-power pressure

(continued)

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low Coincident with:	1,2,3	3 per SG	M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 16.4%	≥ 17.0%
1) Vessel ΔT equivalent to power ≤ 50% RTP	1,2,3	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP
With a time delay (Ts) if one S/G is affected or					≤ 1.01 Ts (Note 1, (Page 3.3-40)	≤ Ts (Note 1, (Page 3.3-40)
A time delay (Tm) if two or more S/G's are affected <u>OR</u>					≤ 1.01 Tm (Note 1, (Page 3.3-40)	≤ Tm (Note 1, (Page 3.3-40)
2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2,3	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input ≤ 50% RTP

(continued)

Table 3.3.2-1 (page 5 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
c. Safety Injection	Refer to Function 1 (safety Injection) for all initiation functions and requirements					
d. Loss of Offsite Power	1,2,3	4 per bus	F	Refer to Function 4 of Table 3.3.5-1 for SRs and Allowable Values		
e. Trip of all Main Feedwater Pumps	1,2	1 per pump	J	SR 3.3.2.8 SR 3.3.2.9 SR 3.3.2.10	≥ 48 psig	≥ 50 psig
f. Motor-Driven Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure —Low	1,2,3	3	F	SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	A) ≥ 0.5 psig B) ≥ 1.33 psig	A) ≥ 1.2 psig B) ≥ 2.0 psig
g. Turbine-driven AFW Pump Suction Train A and B Transfer on Suction Pressure--Low	1,2,3	2/train, 2 trains	F	SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	A) ≥ 11.6 psig B) ≥ 12.2 psig	A) ≥ 12.8 psig B) ≥ 13.5 psig
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA

(continued)

(g) Setpoint verification not required.

Table 3.3.2-1 (page 6 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
7. Automatic Switchover to Containment Sump (continued)						
b. Refueling Water Storage Tank (RWST) Level – Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 155.6 inches from Tank Base	≥ 158 inches from Tank Base
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
and						
Coincident with Containment Sump Level – High	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 37.7 in. above el. 702.8 ft	≥ 38.2 in. above el. 702.8 ft
8. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	F	SR 3.3.2.11	NA	NA
b. Pressurizer Pressure, P-11						
(1) Unblock (Auto Reset of SI Block)	1,2,3	3	L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9	≤ 1975.2 psig	≤ 1970 psig
(2) Enable Manual Block of SI	1,2,3	3	L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9	≥1956.8 psig	≥ 1962 psig

Table 3.3.3-1 (page 2 of 2)
Post Accident Monitoring Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS/TRAINS	CONDITION REFERENCED FROM REQUIRED ACTION E.1
26.	Auxiliary Building Passive Sump Level[j]	1,2,3	2	F

(a) Below the P-10 (Power Range Neutron Flux) interlocks.
(b) Above the P-6 (Intermediate Range Neutron Flux) interlocks.
(c) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(d) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, pressure relief valve, or check valve with flow through the valve secured.
(e) A channel consists of two core exit thermocouples (CETs).
(f) The ICCM provides these functions on a plasma display.
(g) Regulatory Guide 1.97, non-Type A, Category 1 Variables.
(h) This function is displayed on the ICCM plasma display and digital panel meters.
(i) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.
(j) Watts Bar specific (not required by Regulatory Guide 1.97) non-Type A Category 1 variable.

Table 3.3.4-1 (page 1 of 1)
Remote Shutdown System Instrumentation and Controls

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
1. Reactivity Control	
a. Source Range Neutron Flux	1
b. Reactor Trip Breaker Position	1 per trip breaker
2. Reactor Coolant System (RCS) Pressure Control	
a. Pressurizer Pressure or RCS Wide Range Pressure	1
b. Pressurizer Power Operated Relief Valve (PORV) Control or Pressurizer Block Valve Control	1 per relief path
c. Pressurizer Heater Control	1
3. RCS Inventory Control	
a. Pressurizer Level Indication	1
b. Charging Flow Control	1
c. Reactor Vessel Head Vent Valves Control or PORVs and Block Valve Control	2 per flow path
4. Decay Heat Removal via Steam Generators (SGs)	
a. RCS Hot Leg Temperature Indication	1 per loop
b. AFW Controls	1
c. SG Pressure Indication and Control	1 per SG
d. SG Level Indication or AFW Flow Indication	1 per SG
5. Decay Heat Removal via RHR System	
a. RHR FLow Control	1
b. RHR Temperature Indication	1

Table 3.3.6-1 (page 1 of 1)
Containment Vent Isolation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Initiation	2	SR 3.3.6.6	NA
2. Automatic Actuation Logic and Actuation Relays	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Purge Exhaust Radiation Monitors	2	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	≤ 8.41E-02 μCi/cc (8.41E+04 cpm)
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Initiation	2 trains	SR 3.3.7.3	NA
2. Control Room Radiation Control Room Air Intakes	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.4	≤ 5.77E-04 µC/cc (20199 cpm)
3. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ 2214 psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ≤ 593.5°F.	12 hours
SR 3.4.1.3	-------------------NOTE-------------------- Not required to be performed until 12 hours after the installed flow instrumentation has been calibrated to the precision heat balance of SR 3.4.1.4. --- Verify RCS total flow rate is ≥ 397,000 gpm (process computer) or 401,000 gpm (control board indication).	12 hours
SR 3.4.1.4	-------------------NOTE-------------------- Not required to be performed until 24 hours after ≥ 90% RTP. --- Verify by precision heat balance that RCS total flow rate is ≥ 397,000 gpm.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops — MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (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 loop 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 RCS boron concentration; and
   c. No draining operations to further reduce the RCS water volume are permitted.
2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

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

While this LCO is not met, entry into MODE 5, Loops Not Filled is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR loop inoperable.	A.1	Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Cold Overpressure Mitigation System (COMS)

LCO 3.4.12 A COMS System shall be OPERABLE with a maximum of one charging pump and no safety injection pump capable of injecting into the RCS and the accumulators isolated and either a or b below.

a. Two RCS relief valves, as follows:

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

   2. One PORV with a lift setting within the limits specified in the PTLR and the RHR suction relief valve with a setpoint ≥ 436.5 psig and ≤ 463.5 psig.

b. The RCS depressurized and an RCS vent capable of relieving > 475 gpm water flow.

APPLICABILITY: MODES 4 and 5,
MODE 6 when the reactor vessel head is on.

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

1. While this LCO is not met, entry into the Applicability of the LCO is not permitted.

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

3. For the purposes of making the required safety injection pumps and charging pumps inoperable, the following time is permitted. Up to 4 hours after entering MODE 4 from MODE 3, or prior to decreasing temperature on any RCS loop to below 325°F, whichever occurs first.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1	Increase RCS cold leg temperature to > 350°F.	12 hours
	<u>OR</u>		
	D.2	Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
E. One required RCS relief valve inoperable in MODE 4.	E.1	Restore required RCS relief valve to OPERABLE status.	7 days
F. One required RCS relief valve inoperable in MODE 5 or 6.	F.1	Restore required RCS relief valve to OPERABLE status.	24 hours
G. Two required RCS relief valves inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A, B, D, E, or F not met. <u>OR</u> COMS inoperable for any reason other than Condition A, B, C, D, E, or F.	G.1	Depressurize RCS and establish RCS vent.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify no safety injection pumps are capable of injecting into the RCS.	Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F. __AND__ 12 hours thereafter
SR 3.4.12.2	Verify a maximum of one charging pump is capable of injecting into the RCS.	Within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more RCS cold legs decreasing below 325°F. __AND__ 12 hours thereafter
SR 3.4.12.3	Verify each accumulator is isolated.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.4	-----NOTE----- Only required to be performed when complying with LCO 3.4.12.b. ---------- Verify RCS vent open.	12 hours for unlocked open vent paths AND 31 days for locked open vent paths
SR 3.4.12.5	Verify PORV block valve is open for each required PORV.	72 hours
SR 3.4.12.6	Verify both RHR suction isolation valves are locked open with operator power removed for the required RHR suction relief valve.	31 days
SR 3.4.12.7	-----NOTE----- Not required to be met until 12 hours after decreasing RCS cold leg temperature to $\leq 350°F$. ---------- Perform a COT on each required PORV, excluding actuation.	31 days
SR 3.4.12.8	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	-------------------NOTE-------------------- Required to be performed once per 184 days within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS – Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

---NOTES---

1. In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

2. The provisions of Specifications LCO 3.0.4 and SR 3.0.4 are not applicable for entry into MODE 3 for the safety injection pumps and charging pumps declared inoperable pursuant to Specification 3.4.12 for up to four hours or until the temperature of all the RCS cold legs exceeds 375°F, whichever occurs first.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. <u>AND</u> At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.9.3	Verify each EGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.6.9.4	Verify each EGTS train produces a flow rate $\geq$ 3600 and $\leq$ 4400 cfm within 20 seconds from the initiation of a Containment Isolation Phase A signal.	18 months on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.13.4	Verify, by peel test on three specimens for each replacement seal repair location, that the length of peel for at least two of the test specimens is less than or equal to 1 inch.	Prior to initial fuel loading for joints made prior to fuel loading <u>AND</u> 18 months for the first two refueling outages after fabrication of any joint <u>AND</u> 18 months thereafter for a fabricated splice joint, if any of the three test specimens peel length is > 1/2 inch <u>OR</u> 36 months thereafter for a fabricated splice joint, if all three associated test specimens peel length is ≤ 1/2 inch

(continued)

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Applicable Power in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	APPLICABLE POWER (% RTP)
5	≤ 100
4	≤ 59
3	≤ 42
2	≤ 26

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.2.1	-------------------NOTE-------------------- Required to be performed in MODE 3. --- Verify closure time of each MSIV is ≤ 5.0 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program or 18 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two ABGTS trains inoperable during movement of irradiated fuel assemblies in the fuel handling area.	D.1 Suspend movement of irradiated fuel assemblies in the fuel handling area.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each ABGTS train for $\geq$ 10 continuous hours with the heaters operating.	31 days
SR 3.7.12.2	Perform required ABGTS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.12.3	Verify each ABGTS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.12.4	Verify one ABGTS train can maintain a pressure between -0.25 and -0.5 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate $\geq$ 9300 and $\leq$ 9900 cfm.	18 months on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.7	Verify for an actual or simulated loss of offsite power each DG starts from standby condition and achieves in ≤ 10 seconds, voltage ≥ 6800 V, and frequency ≥ 58.8 Hz. Verify after DG fast start from standby conditions that the DG achieves steady state voltage ≥ 6800 V and ≤ 7260 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	184 days
SR 3.8.1.8	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. -- Verify automatic and manual transfer of each 6.9 kV shutdown board power supply from the normal offsite circuit to each alternate offsite circuit.	 18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. --	
	Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and: a. In ≤ 10 seconds after auto-start and during tests, achieves voltage ≥ 6800 V and frequency ≥ 58.8 Hz; b. After DG fast start from standby conditions the DG achieves steady state voltage ≥ 6800 V and ≤ 7260 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz. c. Operates for ≥ 5 minutes; d. Permanently connected loads remain energized from the offsite power system; and e. Emergency loads are energized from the offsite power system.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.15	------------------NOTE------------------ This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 2 hours loaded ≥ 3960 kW and ≤ 4400 kW. Momentary transients outside of load range do not invalidate this test. -- Verify each DG starts and achieves, in ≤ 10 seconds, voltage ≥ 6800 V, and frequency ≥ 58.8 Hz. Verify after DG fast start from standby conditions that the DG achieves steady state voltage ≥ 6800 V and ≤ 7260 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	18 months
SR 3.8.1.16	------------------NOTE------------------ This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. -- Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.21	Verify when started simultaneously from standby condition, each DG achieves, in ≤ 10 seconds, voltage ≥ 6800 V and frequency ≥ 58.8 Hz. Verify after DG fast start from standby conditions that the DG achieves steady state voltage ≥ 6800 V and ≤ 7260 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	10 years

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with starting air receiver pressure < 190 psig and ≥ 170 psig.	E.1 Restore starting air receiver pressure to ≥ 190 psig.	48 hours
F. Required Action and associated Completion Time not met. <u>OR</u> One or more DGs diesel fuel oil, lube oil, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E.	F.1 Declare associated DG inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.3.1	Verify each 7 day fuel oil storage tank contains ≥ 56,754 gal of fuel.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.2	Verify lubricating oil inventory is ≥ 287 gal per engine.	31 days
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG air start receiver pressure is ≥ 190 psig.	31 days
SR 3.8.3.5	Check for and remove accumulated water from each of the four interconnected tanks which constitute the 7 day fuel oil storage tank.	31 days
SR 3.8.3.6	Perform a visual inspection for leaks in the exposed fuel oil system piping while the DG is running.	18 months
SR 3.8.3.7	For each of the four interconnected tanks which constitute the 7 day fuel oil storage tank: a. Drain the fuel oil; b. Remove the sediment; and c. Clean the tank.	10 years

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMIT FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[a]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[b][c]	≥ 1.200	≥ 1.195 <u>AND</u> Average of all connected cells > 1.205	Not more than 0.020 below average of all connected cells <u>AND</u> Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum level during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge for vital batteries and < 1.0 amp for DG batteries.

(c) A battery charging current of < 2 amps when on float charge for vital batteries and < 1.0 amp for DG batteries is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 31 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 31 day allowance.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

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

APPLICABILITY: MODE 6.

-----------------------------NOTE-----------------------------
With the RCS boron concentration specified in the COLR for MODE 6 not met, entry into MODE 6 is not permitted.

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

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

---------------------------NOTE---------------------------
While this LCO is not met, entry into MODE 6 is not permitted.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each unborated water source isolation valve.

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 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Initiate action to secure valve in closed position.	Immediately
	<u>AND</u>	
	A.3 Perform SR 3.9.1.1.	4 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.

---------------------------NOTE---------------------------
Prior to initial criticality, only one RHR loop needs to be OPERABLE and in operation and the required RHR loop may be removed from operation for $\leq$ 1 hour per 8-hour period provided no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration.

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

---------------------------NOTE---------------------------
While this LCO is not met, entry into MODE 6 with water level < 23 ft above the top of the reactor vessel flange is not permitted.

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. __OR__ A.2 Initiate action to establish $\geq$ 23 ft of water above the top of reactor vessel flange.	Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR loop in operation.	B.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	<u>AND</u>		
	B.2	Initiate action to restore one RHR loop to operation.	Immediately
	<u>AND</u>		
	B.3	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	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 $\geq$ 2000 gpm.	12 hours
SR 3.9.6.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days

3.9 REFUELING OPERATIONS

3.9.9 Spent Fuel Pool Boron Concentration

LCO 3.9.9 Boron concentration of the spent fuel pool shall be ≥ 2000 ppm.

APPLICABILITY: During fuel movement in the flooded spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration < 2000 ppm.	A.1 Suspend fuel movement.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.9.1 Verify boron concentration in the spent fuel pool is ≥ 2000 ppm.	Prior to movement of fuel in the spent fuel pool <u>AND</u> 72 hours thereafter

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 3.50 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

c. A nominal 10.72 inch center to center distance between fuel assemblies placed in the high density fuel storage racks.

d. With the spent fuel storage pool flooded, fuel assemblies must not be loaded in cells on the pool periphery or loaded in a configuration which would contain fuel assemblies in cells which are face adjacent.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.3 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 747′ - 1 1/2".

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 484 fuel assemblies.

5.2 Organization

5.2.2 Unit Staff (continued)

of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;
4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized in advance by the Plant Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The Operations Manager shall hold or have held an SRO license on a similar unit.

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Operations Supervisor (SOS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on shift (Generic Letter 86-04 dated 02/13/86).

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

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

LCO 3.1.4 Moderator Temperature Coefficient
LCO 3.1.6 Shutdown Bank Insertion Limit
LCO 3.1.7 Control Bank Insertion Limits
LCO 3.2.1 Heat Flux Hot Channel Factor
LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3 Axial Flux Difference
LCO 3.9.1 Boron Concentration

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

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalphy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

2. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

3. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).(Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

(continued)

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
i	0	11/94
ii	0	11/94
iii	0	05/95
iv	0	11/94
v	0	11/94
vi	0	11/94
vii	0	11/94
viii	0	05/95
ix	0	05/95
x	0	11/94
xi	0	05/95
xii	0	05/95
xiii	0	05/95
xiv	0	05/95
xv	0	05/95
xvi	0	05/95
xvii	0	05/95
xviii	0	05/95
xix	0	05/95
xx	0	05/95
xxi	0	05/95
xxii	0	05/95
xxiii	0	05/95
xxiv	0	05/95

PAGE	REVISION	DATE
B 2.0-1	0	11/94
B 2.0-2	0	11/94
B 2.0-3	0	11/94
B 2.0-4	0	11/94
B 2.0-5	0	11/94
B 2.0-6	0	11/94
B 2.0-7	0	11/94
B 2.0-8	0	05/95
B 2.0-9	0	05/95
B 2.0-10	0	11/94
B 2.0-11	0	11/94
B 2.0-12	0	11/94
B 3.0-1	0	11/94
B 3.0-2	0	11/94
B 3.0-3	0	11/94
B 3.0-4	0	11/94
B 3.0-5	0	11/94
B 3.0-6	0	05/95
B 3.0-7	0	05/95
B 3.0-8	0	05/95

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.2-17	0	11/94
B 3.2-18	0	11/94
B 3.2-19	0	11/94
B 3.2-20	0	11/94
B 3.2-21	0	11/94
B 3.2-22	0	11/94
B 3.2-23	0	11/94
B 3.2-24	0	11/94
B 3.2-25	0	11/94
B 3.2-26	0	11/94
B 3.2-27	0	11/94
B 3.2-28	0	11/94
B 3.2-29	0	11/94
B 3.2-30	0	11/94
B 3.3-1	0	11/94
B 3.3-2	0	11/94
B 3.3-3	0	11/94
B 3.3-4	0	11/94
B 3.3-5	0	11/94
B 3.3-6	0	11/94
B 3.3-7	0	11/94
B 3.3-8	0	11/94
B 3.3-9	0	11/94
B 3.3-10	0	11/94
B 3.3-11	0	11/94
B 3.3-12	0	11/94
B 3.3-13	0	11/94
B 3.3-14	0	05/95
B 3.3-15	0	11/94
B 3.3-16	0	11/94
B 3.3-17	0	11/94
B 3.3-18	0	11/94
B 3.3-19	0	11/94
B 3.3-20	0	11/94
B 3.3-21	0	11/94
B 3.3-22	0	11/94
B 3.3-23	0	11/94
B 3.3-24	0	11/94
B 3.3-25	0	11/94
B 3.3-26	0	11/94
B 3.3-27	0	11/94
B 3.3-28	0	11/94
B 3.3-29	0	11/94
B 3.3-30	0	11/94
B 3.3-31	0	11/94
B 3.3-32	0	11/94

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.4-44	0	11/94
B 3.4-45	0	11/94
B 3.4-46	0	11/94
B 3.4-47	0	11/94
B 3.4-48	0	11/94
B 3.4-49	0	11/94
B 3.4-50	0	11/94
B 3.4-51	0	11/94
B 3.4-52	0	11/94
B 3.4-53	0	11/94
B 3.4-54	0	11/94
B 3.4-55	0	11/94
B 3.4-56	0	11/94
B 3.4-57	0	11/94
B 3.4-58	0	11/94
B 3.4-59	0	11/94
B 3.4-60	0	11/94
B 3.4-61	0	11/94
B 3.4-62	0	11/94
B 3.4-63	0	11/94
B 3.4-64	0	11/94
B 3.4-65	0	11/94
B 3.4-66	0	11/94
B 3.4-67	0	05/95
B 3.4-68	0	11/94
B 3.4-69	0	11/94
B 3.4-70	0	05/95
B 3.4-71	0	11/94
B 3.4-72	0	11/94
B 3.4-73	0	11/94
B 3.4-74	0	05/95
B 3.4-75	0	11/94
B 3.4-76	0	11/94
B 3.4-77	0	11/94
B 3.4-78	0	11/94
B 3.4-79	0	11/94
B 3.4-80	0	11/94
B 3.4-81	0	11/94
B 3.4-82	0	11/94
B 3.4-83	0	11/94
B 3.4-84	0	11/94
B 3.4-85	0	11/94
B 3.4-86	0	11/94
B 3.4-87	0	11/94
B 3.4-88	0	11/94
B 3.4-89	0	11/94
B 3.4-90	0	11/94

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.4-91	0	11/94
B 3.4-92	0	11/94
B 3.4-93	0	11/94
B 3.4-94	0	11/94
B 3.4-95	0	11/94
B 3.4-96	0	11/94
B 3.4-97	0	11/94
B 3.4-98	0	05/95
B 3.4-99	0	11/94
B 3.4-100	0	11/94
B 3.4-101	0	11/94
B 3.5-1	0	11/94
B 3.5-2	0	11/94
B 3.5-3	0	11/94
B 3.5-4	0	11/94
B 3.5-5	0	05/95
B 3.5-6	0	11/94
B 3.5-7	0	11/94
B 3.5-8	0	05/95
B 3.5-9	0	11/94
B 3.5-10	0	11/94
B 3.5-11	0	11/94
B 3.5-12	0	11/94
B 3.5-13	0	11/94
B 3.5-14	0	11/94
B 3.5-15	0	05/95
B 3.5-16	0	11/94
B 3.5-17	0	11/94
B 3.5-18	0	11/94
B 3.5-19	0	11/94
B 3.5-20	0	11/94
B 3.5-21	0	11/94
B 3.5-22	0	11/94
B 3.5-23	0	11/94
B 3.5-24	0	11/94
B 3.5-25	0	11/94
B 3.5-26	0	11/94
B 3.5-27	0	11/94
B 3.5-28	0	11/94
B 3.5-29	0	11/94
B 3.5-30	0	11/94
B 3.5-31	0	11/94
B 3.5-32	0	11/94
B 3.5-33	0	11/94
B 3.5-34	0	11/94
B 3.6-1	0	11/94
B 3.6-2	0	11/94
B 3.6-3	0	11/94

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.7-1	0	11/94
B 3.7-2	0	11/94
B 3.7-3	0	05/95
B 3.7-4	0	05/95
B 3.7-5	0	11/94
B 3.7-6	0	11/94
B 3.7-7	0	11/94
B 3.7-8	0	11/94
B 3.7-9	0	11/94
B 3.7-10	0	11/94
B 3.7-11	0	11/94
B 3.7-12	0	11/94
B 3.7-13	0	11/94
B 3.7-14	0	11/94
B 3.7-15	0	11/94
B 3.7-16	0	11/94
B 3.7-17	0	11/94
B 3.7-18	0	11/94
B 3.7-19	0	11/94
B 3.7-20	0	05/95
B 3.7-21	0	11/94
B 3.7-22	0	11/94
B 3.7-23	0	11/94
B 3.7-24	0	11/94
B 3.7-25	0	11/94
B 3.7-26	0	11/94
B 3.7-27	0	11/94
B 3.7-28	0	11/94
B 3.7-29	0	11/94
B 3.7-30	0	11/94
B 3.7-31	0	11/94
B 3.7-32	0	11/94
B 3.7-33	0	11/94
B 3.7-34	0	11/94
B 3.7-35	0	11/94
B 3.7-36	0	11/94
B 3.7-37	0	11/94
B 3.7-38	0	05/95
B 3.7-39	0	11/94
B 3.7-40	0	11/94
B 3.7-41	0	11/94
B 3.7-42	0	11/94
B 3.7-43	0	11/94
B 3.7-44	0	11/94
B 3.7-45	0	11/94
B 3.7-46	0	05/95
B 3.7-47	0	11/94

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.7-48	0	11/94
B 3.7-49	0	11/94
B 3.7-50	0	11/94
B 3.7-51	0	11/94
B 3.7-52	0	11/94
B 3.7-53	0	11/94
B 3.7-54	0	11/94
B 3.7-55	0	11/94
B 3.7-56	0	11/94
B 3.7-57	0	11/94
B 3.7-58	0	11/94
B 3.7-59	0	11/94
B 3.7-60	0	11/94
B 3.7-61	0	05/95
B 3.7-62	0	11/94
B 3.7-63	0	11/94
B 3.7-64	0	11/94
B 3.7-65	0	11/94
B 3.7-66	0	05/95
B 3.7-67	0	11/94
B 3.7-68	0	11/94
B 3.7-69	0	11/94
B 3.7-70	0	11/94
B 3.7-71	0	11/94
B 3.7-72	0	11/94
B 3.7-73	0	11/94
B 3.7-74	0	11/94
B 3.8-1	0	11/94
B 3.8-2	0	11/94
B 3.8-3	0	11/94
B 3.8-4	0	11/94
B 3.8-5	0	11/94
B 3.8-6	0	11/94
B 3.8-7	0	11/94
B 3.8-8	0	11/94
B 3.8-9	0	11/94
B 3.8-10	0	11/94
B 3.8-11	0	11/94
B 3.8-12	0	11/94
B 3.8-13	0	11/94
B 3.8-14	0	11/94
B 3.8-15	0	11/94
B 3.8-16	0	11/94
B 3.8-17	0	11/94
B 3.8-18	0	11/94
B 3.8-19	0	11/94
B 3.8-20	0	11/94
B 3.8-21	0	11/94

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.8-69	0	11/94
B 3.8-70	0	11/94
B 3.8-71	0	11/94
B 3.8-72	0	11/94
B 3.8-73	0	11/94
B 3.8-74	0	11/94
B 3.8-75	0	11/94
B 3.8-76	0	11/94
B 3.8-77	0	11/94
B 3.8-78	0	11/94
B 3.8-79	0	11/94
B 3.8-80	0	05/95
B 3.8-81	0	11/94
B 3.8-82	0	11/94
B 3.8-83	0	11/94
B 3.8-84	0	11/94
B 3.8-85	0	11/94
B 3.8-86	0	11/94
B 3.8-87	0	11/94
B 3.8-88	0	11/94
B 3.8-89	0	11/94
B 3.8-90	0	11/94
B 3.8-91	0	11/94
B 3.8-92	0	11/94
B 3.8-93	0	11/94
B 3.8-94	0	11/94
B 3.8-95	0	11/94
B 3.8-96	0	11/94
B 3.8-97	0	11/94
B 3.8-98	0	11/94
B 3.8-99	0	11/94
B 3.8-100	0	11/94
B 3.8-101	0	11/94
B 3.8-102	0	05/95
B 3.9-1	0	05/95
B 3.9-2	0	05/95
B 3.9-3	0	05/95
B 3.9-4	0	05/95
B 3.9-5	0	11/94
B 3.9-6	0	05/95
B 3.9-7	0	11/94
B 3.9-8	0	05/95
B 3.9-9	0	11/94
B 3.9-10	0	11/94
B 3.9-11	0	11/94
B 3.9-12	0	11/94
B 3.9-13	0	11/94
B 3.9-14	0	11/94
B 3.9-15	0	11/94
B 3.9-16	0	11/94
B 3.9-17	0	11/94
B 3.9-18	0	11/94

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.9-19	0	11/94
B 3.9-20	0	11/94
B 3.9-21	0	11/94
B 3.9-22	0	05/95
B 3.9-23	0	05/95
B 3.9-24	0	05/95
B 3.9-25	0	11/94
B 3.9-26	0	11/94
B 3.9-27	0	11/94
B 3.9-28	0	11/94
B 3.9-29	0	11/94
B 3.9-30	0	11/94
B 3.9-31	0	11/94
B 3.9-32	0	11/94
B 3.9-33	0	05/95
B 3.9-34	0	05/95

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs (Ref. 8), provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Pressurizer power operated relief valves (PORVs);

b. Steam line power operated relief valve (PORV);

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel, piping, valves, and fittings under the ASME Code, Section III, is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2735 psig.

(continued)

BASES

LCO 3.0.4 (continued)

practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

(continued)

BASES

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

(continued)

BASES

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.7.2.18, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations,

(continued)

BASES

LCO 3.0.6 (continued)

remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the plant. These special tests and operations are necessary to demonstrate select plant performance characteristics, to perform special maintenance activities, and to perform special evolutions.

Test Exception LCOs 3.1.9, 3.1.10 and 3.4.17 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

BASES

SR 3.0.4 (continued)

safe operation of the unit. However, in certain circumstances failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillances(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not

(continued)

BASES

SR 3.0.4 (continued)	required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.
	SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

BASES

APPLICABLE SAFETY ANALYSES (continued)

rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

APPLICABILITY

In MODE 2 with k_{eff} < 1.0 and in MODES 3 and 4, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 5, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM) – $T_{avg} \leq$ 200°F." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6 and LCO 3.1.7.

ACTIONS

A Note to the ACTIONS precludes use of LCO 3.0.4 to permit decreasing temperature while not meeting the SDM.

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that

(continued)

BASES (continued)

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

APPLICABILITY

In MODE 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 2 with K_{eff} < 1.0 and MODES 3 and 4, the SDM requirements are given in LCO 3.1.1, "SHUTDOWN MARGIN (SDM) − T_{avg} > 200°F." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODE 1 and MODE 2 with $K_{eff} \geq 1.0$, SDM is ensured by complying with LCO 3.1.6 and LCO 3.1.7.

ACTIONS

A Note to the ACTIONS precludes use of LCO 3.0.4 to permit decreasing temperature while not meeting the SDM.

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a concentrated solution, such as that normally found in the boric acid storage tank or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/k must be recovered and a boration flow rate of

(continued)

BASES

ACTIONS

A.1 (continued)

35 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 35 gpm and 6120 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

In MODE 5, the SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Design isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM. This allows time enough for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides backup protection to the Power Range Neutron Flux—Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux—High Setpoint trip provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM.

(continued)

BASES

ACTIONS

N.1 and N.2 (continued)

other RTS trip Functions provide core protection below the P-8 setpoint. The 6 hours allowed to restore the channel to OPERABLE status or place in trip and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 4 hour time limit is justified in Reference 7.

O.1 and O.2

Condition O applies to Turbine Trip on Low Fluid Oil Pressure. With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the tripped condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 7.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

P.1 and P.2

Condition P applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed

(continued)

BASES

ACTIONS

U.1.1, U.1.2, and U.2 (continued)

place the plant in MODE 3. Six hours is a reasonable time, based on operating experience, to place the plant in MODE 3 from MODE 1 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 4 hour time limit is justified in Reference 7.

V.1 and V.2

Condition V applies to the Vessel ΔT Equivalent to Power reactor trip Function.

Failure of the vessel ΔT channel input (failure of more than one T_H RTD or failure of both T_c RTDs) affects the TTD calculation for a protection set. This results in the requirement that the operator adjust the threshold power level for zero seconds time delay from 50% RTP to 0% RTP, through the Man Machine Interface.

If the inoperable channel cannot be restored or the threshold power level for zero seconds time delay adjusted within the specified Completion Time, the plant must be placed in a MODE where these Functions are not required to be OPERABLE. An additional 6 hours is allowed to place the plant in MODE 3. Six hours is a reasonable time, based on operating experience, to place the plant in MODE 3 from MODE 1 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 4 hour time limit is justified in Reference 7.

(continued)

BASES

ACTIONS

X.1 and X.2 (continued)

- Reactor Coolant Flow-Low (Two Loops).

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint and below the P-8 setpoint. These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition X.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 4 hour time limit is justified in Reference 7.

Y.1 and Y.2

Condition Y applies to the Turbine Trip on Stop Valve Closure. With one, two or three channels inoperable, the inoperable channels must be placed in the trip condition within 6 hours. Since all the valves must be tripped (not fully open), in order for the reactor trip signal to be generated, it is acceptable to place more than one Turbine Stop Valve Closure channel in the trip condition. With one or more channels in the trip condition, a partial reactor trip condition exists. All of the remaining Turbine Stop Valve channels are required to actuate in order to initiate a reactor trip. If a channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced to below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place an inoperable channel in the trip condition and the 4 hours allowed for reducing power are justified in Reference 7.

(continued)

BASES

ACTIONS (continued)

Z.1

With two RTS trains inoperable, no automatic capability is available to shutdown the reactor, and immediate plant shutdown in accordance with the LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV. The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

The protection Functions associated with the EAGLE-21™ Process Protection System have an installed bypass capability, and may be tested in either the trip or bypass mode, as approved in Reference 7. When testing is performed in the bypass mode, the SSPS input relays are not operated, as justified in Reference 9. The input relays are checked during the CHANNEL CALIBRATION every 18 months.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days except for source range instrument channels which are every 31 days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of References 6 and 7.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for greater than 4 hours, this Surveillance must be performed within 4 hours after entry into MODE 3.

The Frequency of 92 days is justified in Reference 7, except for Function 13. The justification for Function 13 is provided in Reference 9.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days (31 days for source and intermediate range instrument channels) prior to reactor startup and 4 hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.8 (continued)

performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "4 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days (31 days for source and intermediate range instrument channels) thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range-low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 92 days, as justified in Reference 7.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the Watts Bar setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of sensor/transmitter drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable. For channels with a trip time delay (TTD), this test shall include verification that the TTD coefficients are adjusted correctly.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, 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 power range neutron detectors consists of a normalization of the detectors based on a power calorimetric performed above 15% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 (continued)

SR 3.3.1.15 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

REFERENCES

1. Watts Bar FSAR, Section 6.0, "Engineered Safety Features."
2. Watts Bar FSAR, Section 7.0, "Instrumentation and Controls."
3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.
5. 10 CFR Part 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."
6. WCAP-12096, Rev. 6, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," May 1994.
7. WCAP-10271-P-A, Supplement 1, and Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," May 1986 and June 1990.
8. Watts Bar Technical Requirements Manual, Section 3.3.1, "Reactor Trip System Response Times."
9. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar.
10. ISA-DS-67.04, 1982, "Setpoint for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants."

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Automatic Switchover to Containment Sump (continued)

Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

a. Automatic Switchover to Containment Sump-Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Low Coincident With Safety Injection and Coincident With Containment Sump Level-High

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low low level in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability.

The RWST – Low Low Trip Setpoint is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Low Coincident With Safety Injection and Coincident With Containment Sump Level-High (continued)

This setpoint will also ensure that enough borated water is injected to maintain the reactor shut down. The limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction.

The transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Automatic switchover occurs only if the RWST low low level signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. The automatic switchover Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

Additional protection from spurious switchover is provided by requiring a Containment Sump Level – High signal as well as RWST Level – Low Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level – High signal must be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Low Coincident With Safety Injection and Coincident With Containment Sump Level-High (continued)

present, in addition to the SI signal and the RWST Level – Low Low signal, to transfer the suctions of the RHR pumps to the containment sump. The containment sump is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability. The containment sump level Trip Setpoint is selected to ensure enough borated water is injected to ensure the reactor remains shut down. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction. The transmitters are located inside containment and thus possibly experience adverse environmental conditions. Therefore, the Trip Setpoint reflects the inclusion of both steady state and environmental instrument uncertainties.

These Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The protection Functions associated with the EAGLE-21™ Process Protection System have an installed bypass capability, and may be tested in either the trip or bypass mode, as approved in Reference 7. When testing is performed in the bypass mode, the SSPS input relays are not operated, as justified in Reference 10. The input relays are checked during the CHANNEL CALIBRATION every 18 months.

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.4 (continued)

The Frequency of 92 days is justified in Reference 7, except for Function 7. The Frequency for Function 7 is justified in Reference 10.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 92 days. The Frequency is adequate, based on industry operating experience, considering instrument reliability and operating history data.

This SR is modified by a Note, which states that performance of this test is not required for those relays tested by SR 3.3.2.7.

SR 3.3.2.6

SR 3.3.2.6 is the performance of a TADOT every 92 days. This test is a check of the Loss of Offsite Power (Function 6.d), AFW Pump Suction Transfer on Suction Pressure—Low for motor driven and turbine driven pumps (Functions 6.f and 6.g respectively), and Turbine Trip and Feedwater Isolation - Main Steam Valve Vault Rooms Water Level - High (Function 5.d).

The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. The Frequency is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.9 (continued)

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of sensor/transmitter drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable. For channels with a trip time delay (TTD), this test shall include verification that the TTD coefficients are adjusted correctly.

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in Technical Requirements Manual, Section 3.3.2 (Ref. 8). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of sequential tests such that the entire response time is measured.

ESF RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel.

(continued)

BASES

REFERENCES (continued)

6. WCAP-12096, Rev. 6, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," May 1994.

7. WCAP-10271-P-A, Supplement 1 and Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System." May 1986 and June 1990.

8. Watts Bar Technical Requirements Manual, Section 3.3.2, "Engineered Safety Feature Response Times."

9. TVA Letter to NRC, November 9, 1984, "Request for Exemption of Quarterly Slave Relay Testing, (L44 841109 808)."

10. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar.

11. Westinghouse letter to TVA (WAT-D-8347), September 25, 1990, "Charging/Letdown Isolation Transients" (T33 911231 810).

BASES

LCO

11. Containment Isolation Valve Position (continued)

normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

A Note to the Required Channels states that the Function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, pressure relief valve, or check valve with flow through the valve secured.

12. Containment Radiation (High Range)

Containment Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

Containment radiation level is also used to determine if a loss of reactor coolant or secondary coolant has occurred.

13. Containment Hydrogen Concentration

Hydrogen Monitors, a non-Type A Category 1 variable, are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions.

Hydrogen concentration is also used to determine whether or not to start the hydrogen ignitors and/or recombiners. Containment hydrogen instrumentation consists of two trains on separate power supplies with a range of 0-10% (by volume) hydrogen concentration.

(continued)

BASES

LCO (continued)

17. AFW Valve Status

The status of each AFW swapover to Essential Raw Cooling Water (ERCW) valve is monitored with non-Type A Category 1 indication in the control room. Indication on each valve for fully open or fully closed position is provided. AFW valve status is monitored to give verification to the operator that automatic transfer to ERCW has taken place.

18, 19, 20, 21. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling.

Core exit thermocouples, in conjunction with RCS wide range temperatures, are sufficient to provide indication of radial distribution of the coolant enthalpy rise across representative sections of the core. Core Exit Temperature is used to support determination of whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.

The Inadequate Core Cooling Monitor (ICCM) is used to monitor the core exit thermocouples. There are two isolated systems, with each system monitoring at least four thermocouples per quadrant. The plasma display gives the average quadrant value, the high quadrant value, and the low quadrant value for each quadrant.

Two OPERABLE channels are required in each quadrant to provide adequate indication of coolant temperature rise in representative regions of the core. Two isolated channels of two thermocouples each ensure a single failure will not disable the ability to identify significant temperature gradients.

The incore thermocouple monitoring system described in Reference 4 supports the plant operating procedures.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.4

SR 3.3.3.4 is the performance of a TADOT. This test is performed every 18 months. The test checks operation of the containment isolation valve position indicators and AFW valve position indicators. The Frequency is based on the known reliability of the indicators and has been shown to be acceptable through operating experience.

This SR has been modified by two Notes. Note 1 excludes verification of setpoints for the valve position indicators. Note 2 indicates that this SR is only applicable to Functions 11 and 17, which are the only Functions with valve position indicators.

REFERENCES

1. NUREG-0847, Safety Evaluation Report, Supplement Number 9, June 16, 1992, Section 7.5.2, "Post Accident Monitoring System."

2. Regulatory Guide 1.97, Revision 2, December 1980, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

3. NUREG-0737, "Clarification of TMI Action Plan Requirements," Supplement 1, January 1983.

4. Submittal from John H. Garrity to U.S. Nuclear Regulatory Commission dated January 24, 1992, "Watts Bar Nulcear Plant (WBN) Units 1 and 2 - NUREG 0737, Item II.F.2 - Instrumentation for Detection of Inadequate Core Cooling (ICC) - Proposed License Condition 3 (TAC Numbers M77132 and M77133).

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves or the SG atmospheric dump valves (ADVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) from outside the control room allows extended operation in MODE 3.

If the control room becomes inaccessible, the operators can establish control in the auxiliary control room, and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located in the auxiliary control room. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the remote shutdown control and instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible. Should it be necessary to go to MODE 4, decay heat removal via the Residual Heat Removal (RHR) System is available to support the transition.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The criteria governing the design and specific system requirements of the Remote Shutdown System are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

The Remote Shutdown System is considered an important contributor to the reduction of unit risk to accidents and as such it has been retained in the Technical Specifications as indicated in the NRC Policy Statement.

LCO

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

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

- Core reactivity control (initial and long term);
- RCS pressure control;
- Decay heat removal via the AFW System and the SG safety valves or SG ADVs;
- RCS inventory control via charging flow and Reactor Vessel Head Vent Valves or Pressurizer PORV and Block Valve operation;
- Decay Heat Removal via RHR System;
- Safety support systems though not specifically listed in Table 3.3.4-1, for the above Functions, including service water, component cooling water, reactor containment fan cooler units, auxiliary control air compressors, and onsite power, including the diesel generators are required as discussed in FSAR Section 7.4 (Reference 2).

(continued)

BASES

LCO

2. Automatic Actuation Logic and Actuation Relays (continued)

MODES and specified conditions for the containment vent isolation portion of the SI Function is different and less restrictive than those for the SI role. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Vent Isolation Function is affected, the Conditions applicable to the SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Vent Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Vent Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups and sample pump operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

Only the Allowable Value is specified for the Containment Purge Exhaust Radiation Monitors in the LCO. The Allowable Value is based on expected concentrations for a small break LOCA, which is more restrictive than 10 CFR 100 limits. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.

(continued)

BASES

LCO

2. Control Room Radiation (continued)

For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

Only the Allowable Value is specified for the Control Room Air Intake Radiation Monitors in the LCO. The Allowable Value is based on 10 CFR 50, Appendix A, Criterion 19 exposure limits considering the most limiting accident, which has been determined to be a steam generator tube rupture event. This event is more limiting than a fuel handling accident event or a LOCA. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.

3. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

APPLICABILITY

The CREVS Functions must be OPERABLE in MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies. The Functions must also be OPERABLE in MODES 5 and 6 when required for a waste gas decay tank rupture accident, to ensure a habitable environment for the control room operators.

(continued)

BASES

ACTIONS (continued)

E.1

Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a waste gas decay tank rupture.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREVS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.2 (continued)

potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

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

This SR is modified by a Note which permits delaying performance of the SR on the initial reactor startup until after the installed flow instrumentation has been calibrated to the data obtained in SR 3.4.1.4.

SR 3.4.1.4

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

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

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after $\geq$ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis," Section 15.2, "Normal Operation and Anticipated Transients," and Section 15.3.4, "Complete Loss Of Forced Reactor Coolant Flow."

BASES

LCO (continued)

Note 1 permits all RHR pumps to be de-energized for ≤ 15 minutes when switching from one loop to another. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet 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 that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A Note to the ACTIONS precludes use of LCO 3.0.4 to permit entry into MODE 5, Loops Not Filled without meeting the LCO.

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate

(continued)

BASES

ACTIONS

A.1 (continued)

Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

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

SR 3.4.8.2

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

REFERENCES

None.

BASES

APPLICABILITY (continued)

The Applicability is modified by three Notes. The provisions of LCO 3.0.4 do not preclude entry into the Applicability of this LCO with the LCO not met. Therefore, Note 1 has been added to preclude entry into this LCO while the LCO is not met. Note 2 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions. Note 3 provides time to make the required pumps inoperable since the COMS arming temperature is the same as MODE 3 to MODE 4 transition temperature.

ACTIONS

A.1 and B.1

With two or more charging pumps or any safety injection pumps capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

Required Action B.1 is modified by a Note that permits two charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.

<u>C.1, D.1, and D.2</u>

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 350°F, an accumulator pressure specified in WAT-D-9448 (Ref. 9) cannot exceed the COMS limits if the accumulators are fully injected. Depressurizing the accumulators below the COMS limit from the PTLR also gives this protection.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (continued)

through the pump control switch being placed in pull to lock and at least one valve in the discharge flow path being closed.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment. The additional Frequency for SR 3.4.12.1 and SR 3.4.12.2 is necessary to allow time during the transition from MODE 3 to MODE 4 to make the pumps inoperable.

SR 3.4.12.4

The RCS vent capable of relieving > 475 gpm water flow is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a vent path that cannot be locked.

b. Once every 31 days for a vent path that is locked, sealed, or secured in position. A removed safety or PORV fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can allow varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA) or steam generator tupe rupture (SGTR).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed once per 184 days within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," 1973.

2. Watts Bar FSAR, Section 15.4, "Condition IV - Limiting Faults."

BASES

APPLICABLE SAFETY ANALYSES (continued)	The accumulators satisfy Criterion 3 of the NRC Policy Statement.
LCO	The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated. For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.
APPLICABILITY	In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist. This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F. In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after a 75 gallons (1% volume) increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is ≥ 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA. This design feature still exists, but is no longer required for accident mitigation.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS – Shutdown."

As indicated in Note 1, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room. Note 2 provides relief from the restrictions of LCO 3.0.4 and SR 3.0.4 to allow time to restore the required equipment to OPERABLE status.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by 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."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.8 (continued)

The frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions and therefore, the frequency extensions of SR 3.0.2 may not be applied since the testing is an Appendix J Type C test. This SR simply imposes additional acceptance criteria. Although not a part of L_a, the Shield Building Bypass leakage path combined leakage rate is determined using the 10 CFR 50, Appendix J, Type B and C leakage rates for the applicable barriers.

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis."
2. Watts Bar FSAR, Section 6.2.4.2, "Containment Isolation System Design," and Table 6.2.4-1, "Containment Penetrations and Barriers."
3. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."

BASES

ACTIONS

C.1 and C.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

SR 3.6.6.2

Verifying that each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 4). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

(continued)

BASES

BACKGROUND (continued)

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters are included to reduce the relative humidity of the airstream on systems that operate in high humidity. Continuous operation of each train, for at least 10 hours per month, with heaters on, reduces moisture buildup on their HEPA filters and adsorbers. Cross-over flow ducts are provided between the two trains to allow the active train to draw air through the inactive train and cool the air to keep the charcoal beds on the inactive train from becoming too hot due to absorption of fission products.

The containment annulus vacuum fans maintain the annulus at -5 inches water gauge vacuum during normal operations. During accident conditions, the containment annulus vacuum fans are isolated from the air cleanup portion of the system.

The EGTS reduces the radioactive content in the shield building atmosphere following a DBA. Loss of the EGTS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis.

APPLICABLE SAFETY ANALYSES

The EGTS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) assumes that only one train of the EGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the remaining one train of this filtration system. The amount of fission products available for release from containment is determined for a LOCA.

The safety analysis assumes an initial annulus vacuum pressure of -5.0 inches water gauge prior to the LOCA. The analysis further assumes that upon receipt of a Phase A isolation signal from the RPS, the EGTS fans automatically start and achieve a minimum flow of 3600 cfm

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.10.1 (continued)

excessive vibration can be detected for corrective action. The 92 day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.10.2

Verifying ARS fan motor current with the return air backdraft dampers closed confirms one operating condition of the fan. This test is indicative of overall fan motor performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of 92 days conforms with the testing requirements for similar ESF equipment and considers the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.10.3

Verifying the OPERABILITY of the air return damper to the proper opening torque (Ref. 3) provides assurance that the proper flow path will exist when the fan is started. By applying the correct torque to the damper shaft, the damper operation can be confirmed. The Frequency of 92 days was developed considering the importance of the dampers, their location, physical environment, and probability of failure. Operating experience has also shown this Frequency to be acceptable.

REFERENCES

1. Watts Bar FSAR, Section 6.8, "Air Return Fans."
2. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."
3. System Description N3-30RB-4002.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.13.2 (continued)

resilient materials in the seals must be opened and inspected at least once every 10 years to provide assurance that the seal material has not aged to the point of degraded performance. The Frequency of 10 years is based on the known resiliency of the materials used for seals, the fact that the openings have not been opened (to cause wear), and operating experience that confirms that the seals inspected at this Frequency have been found to be acceptable.

SR 3.6.13.3

Verification, by visual inspection, after each opening of a personnel access door or equipment hatch that it has been closed makes the operator aware of the importance of closing it and thereby provides additional assurance that divider barrier integrity is maintained while in applicable MODES.

SR 3.6.13.4

The divider barrier seal can be field spliced for repair purposes utilizing a cold bond procedure rather than the original field splice technique of vulcanization. However, the cold bond adhesive, which works in conjunction with a bolt array to splice the field joint, could not be heat aged to 40 years plant life prior to acceptability testing. Prolonged exposure to the elevated temperatures required for heat aging the seal material was destructive to the adhesive. The seal material was heat aged to 40 years equivalent age and the entire joint assembly was irradiated to 40 year normal operation plus accident integrated dose. Conducting periodic peel tests on the test specimens provides assurance that the adhesive has not degraded in the containment environment. The Frequencies of 18 months for the first two outages after fabrication of the joint, followed by 18 months if the peel lengths greater than 1/2" and 36 months if the peel length is less than or equal to 1/2" is based upon the original vendor's recommendation which is based upon baseline examination of the strength of the adhesive. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

LCO (continued)

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

APPLICABILITY

In MODE 1 above 40% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below 40% RTP in MODES 1, 2, and 3, only two MSSVs per steam generator are required to be OPERABLE.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

With one or more MSSVs inoperable, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. With an upper MTC limit of 0 $\Delta k/k/°F$, this is accomplished by maintaining THERMAL POWER at or below the power levels specified in Table 3.7.1.1. This ensures that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. The reduced THERMAL POWER level for a reduced steam relieving capacity can be determined by performing a energy balance between the reactor coolant system heat

(continued)

BASES

ACTIONS

A.1 (continued)

generation and the steam relief through the OPERABLE MSSVs, as shown below:

$$\textit{Allowable THERMAL POWER Level (\%)} = 100\frac{4w_s h_{fg}}{QK}$$

where:

w_s = Minimum total steam relief capacity of the OPERABLE MSSVs on any one steam generator, in lbm/sec.

h_{fg} = heat of vaporization at the highest MSSV full-open pressure, in Btu/lbm.

Q = NSSS power rating of the plant (includes reactor coolant pump heat) in MWt.

K = Unit conversion factor: 947.82 Btu/sec/MWt.

Note: The values in Specification 3.7.1 include an allowance for instrument and channel uncertainties to the allowable RTP obtained with this algorithim.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Dump System to the condenser not be available, as discussed in the FSAR, Section 10.3 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ADV line for each of the four steam generators is provided. Each ADV line consists of one ADV and an associated block valve.

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are provided with a pressurized air supply from the auxiliary air compressors that, on a loss of pressure in the normal instrument air supply, automatically supplies backup air to operate the ADVs. The ADVs are also supplied with nitrogen to permit local operation outside the valve rooms.

A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with a DC power source and control air available. In addition, handwheels are provided for local manual operation.

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions. The capacity of the ADVs is sufficient to achieve a cooldown rate of 50°F/hr throughout the entire cooldown to RHR entry conditions with 2 ADVs in service. This permits a uniform cooldown within the capacity of the cooling water supply available in the CST.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling System (CCS)

BASES

BACKGROUND

The CCS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCS also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCS serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Raw Cooling Water (ERCW) System, and thus to the environment.

The CCS is arranged as two independent, full-capacity cooling trains. Separate Train A equipment is provided in each unit, whereas Train B is shared by both units. Train A in unit 1 is served by CCS Hx A and CCS pump 1A-A. Pump 1B-B, which is actually Train B equipment, is also normally aligned to the Train A header in unit 1. However, pump 1B-B can be realigned to Train B on loss of Train A.

Similarly, Train A in unit 2 is served by CCS Hx B and CCS pump 2A-A with support from pump 2B-B.

Train B in both units is served by CCS Hx C. Normally, only CCS pump C-S is aligned to the Train B headers since few nonessential, normally-operating loads are assigned to Train B. However, pumps 1B-B or 2B-B can be realigned to the Train B headers on a loss of the C-S pump. CCS pump 2B-B is aligned with CCS pump C-S and Train 1B during Unit 1 only operation.

Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential components will be manually isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the CCS is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post accident cooldown and shutdown.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

This SR is modified by a Note indicating that the isolation of the ERCW System components or systems may render those components inoperable, but does not affect the OPERABILITY of the ERCW System.

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

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

SR 3.7.8.2

This SR verifies proper automatic operation of the ERCW System valves on an actual or simulated actuation signal. The ERCW System is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	<u>SR 3.7.11.1</u> This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the sizing calculations in the control room. This SR consists of a combination of testing and calculations. This is accomplished by verifying that the system has not degraded. The only measurable parameters that could degrade undetected during normal operation is the system air flow and chilled water flow rate. Verification of these two flow rates will provide assurance that the heat removal capacity of the system is still adequate. The 18 month Frequency is appropriate since significant degradation of the CREATCS is slow and is not expected over this time period.
REFERENCES	1. Watts Bar FSAR, Section 9.4.1, "Control Room Area Ventilation System."

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 8.

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 and -0.5 inches water gauge with respect to atmospheric pressure at a nominal flow rate ≥ 9300 and ≤ 9900 cfm. Periodic testing of ABGTS shall be performed once with one AB general supply fan running and one train of its discharge ABSCE isolation dampers open and once with one purge air supply fan running and one of its suction-side ABSCE isolation dampers open. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 9).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 8.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, "Engineered Safety Feature (ESF) Filter Systems."
2. Watts Bar FSAR, Section 9.4.2, "Fuel Handling Area Ventilation System."
3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (10 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The minimum voltage and frequency stated in the SR are those necessary to ensure the DG can accept DBA loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential to establishing DG OPERABILITY, but a time constraint is not imposed. This is because a typical DG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the SR. In lieu of a time constraint in the SR, WBN will monitor and trend the actual time to reach steady state operation as a means of ensuring there is no voltage regulator or governor degradation which could cause a DG to become inoperable. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d and SR 3.8.1.12.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

The requirement to verify the connection of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

For the purpose of this testing, the DGs shall be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations. The DG engines for WBN have an oil circulation and soakback system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

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

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 10 seconds. The minimum voltage and frequency stated in the SR are those necessary to ensure the DG can accept DBA loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential to establishing DG OPERABILITY, but a time constraint is not imposed. This is because a typical DG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the SR. In lieu of a time constraint in the SR, WBN will monitor and trend the actual time to reach steady state operation as a means of ensuring there is no voltage regulator or governor degradation which could cause a DG to become inoperable. The 10 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1.

The DG engines for WBN have an oil circulation and soakback system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

This SR is modified by a Note to ensure that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least 2 hours at full load conditions

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.15 (continued)

prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test.

SR 3.8.1.16

As required by Regulatory Guide 1.9 (Ref. 3), paragraph C2.2.11, this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1, and takes into consideration plant conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing and the DG will automatically reset to ready to load operation if a LOCA

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

actuation signal is received during operation in the test mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 13), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.18

Under accident and loss of offsite power conditions loads are sequentially connected to the 6.9 kV shutdown board by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.18 (continued)

prevent overloading of the DGs due to high motor starting currents. The load sequence time specified in FSAR Table 8.3-3 ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.19

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.19 (continued)

The Frequency of 18 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

For the purpose of this testing, the DGs shall be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations. The DG engines for WBN have an oil circulation and soakback system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

This SR is modified by a Note. The reason for the Note is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.1.20

This SR verifies that DG availability is not compromised by the idle start circuitry, when in the idle mode of operation, and that an automatic or emergency start signal will disable the idle start circuitry and command the engine to go to full speed. The 18 month frequency is consistent with the expected fuel cycle lengths and is considered sufficient to detect any degradation of the idle start circuitry.

SR 3.8.1.21

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.21 (continued)

proper speed within the specified time when the DGs are started simultaneously. The minimum voltage and frequency stated in the SR are those necessary to ensure the DG can accept DBA loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential to establishing DG OPERABILITY, but a time constraint is not imposed. This is because a typical DG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the SR. In lieu of a time constraint in the SR, WBN will monitor and trend the actual time to reach steady state operation as a means of ensuring there is no voltage regulator or governor degradation which could cause a DG to become inoperable.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1.

For the purpose of this testing, the DGs shall be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations. The DG engines for WBN have an oil circulation and soakback system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

Diesel Generator Test Schedule

The DG test schedule (Table 3.8.1-1) implements the recommendations of Revision 3 to Regulatory Guide 1.9 (Ref. 3). The purpose of this test schedule is to provide timely test data to establish a confidence level associated with the goal to maintain DG reliability > 0.975 per demand.

According to Regulatory Guide 1.9, Revision 3 (Ref. 3), each DG should be tested at least once every 31 days. Whenever a DG has experienced 4 or more valid failures in the last 25 valid tests, the maximum time between tests is reduced to 7 days. Four failures in 25 valid tests is a failure rate of 0.16, or the threshold of acceptable DG performance, and hence may be an early indication of the degradation of DG reliability. When considered in the light of a long history of tests, however, 4 failures in the last 25 valid tests may only be a statistically probable distribution of random events. Increasing the test Frequency will allow for a more timely accumulation of additional test data upon which to base judgment of the reliability of the DG. The increased

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Diesel Generator Test Schedule (continued)

test Frequency must be maintained until seven consecutive, failure free tests have been performed.

The Frequency for accelerated testing is 7 days, but no less than 24 hours. Tests conducted at intervals of less than 24 hours may be credited for compliance with Required Actions. However, for the purpose of re-establishing the normal 31-day Frequency, a successful test at an interval of less than 24 hours should be considered an invalid test and not count towards the 7 consecutive failure free starts, and the consecutive test count is not reset.

A test interval in excess of 7 days (or 31 days as appropriate) constitutes a failure to meet the SRs and results in the associated DG being declared inoperable. It does not, however, constitute a valid test or failure of the DG, and any consecutive test count is not reset.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion (GDC) 17, "Electrical Power Systems."

2. Watts Bar FSAR, Section 8.2, "Offsite Power System,"and Tables 8.3-1 to 8.3-3, "Safety-Related Standby Power sources and Distribution Boards," "Shutdown Board Loads Automatically Tripped Following a Loss of Nuclear Unit and Preferred Power," and "Diesel Generator Load Sequentially Applied Following a Loss of Nuclear Unit and Preferred Power."

3. Regulatory Guide 1.9, Rev. 3, "Selection, Design, Qualification and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," July 1993.

4. Watts Bar FSAR Section 6, "Engineered Safety Features."

5. Watts Bar FSAR, Section 15.4, "Condition IV-Limiting Faults."

6. Regulatory Guide 1.93, Rev. 0, "Availability of Electric Power Sources," December 1974.

7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

(continued)

BASES

REFERENCES (continued)

8. Title 10, Code of Federal Regulations, Part 50, Appendix A, GDC 18, "Inspection and Testing of Electric Power Systems."

9. Regulatory Guide 1.137, Rev. 1 "Fuel Oil Systems for Standby Diesel Generators," October 1979.

BASES

ACTIONS

D.1 (continued)

restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

With starting air receiver pressure < 190 psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is > 170 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit of ≥ 190 psig. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil, lube oil or starting air subsystem not within limits for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The 7 day period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

(continued)

BASES

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 17, "Electric Power System."
2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," U.S. Nuclear Regulatory Commission, March 10, 1971.
3. IEEE-308-1971, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
4. Watts Bar FSAR, Section 8.3.2, "DC Power System."
5. IEEE-485-1983, "Recommended Practices for Sizing Large Lead Storage Batteries for Generating Stations and Substations," Institute of Electrical and Electronic Engineers.
6. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," February 1977, U.S. Nuclear Regulatory Commission.
7. Watts Bar FSAR, Section 15, "Accident Analysis" and Section 6 "Engineered Safety Features."
8. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.
9. IEEE-450-1980, "IEEE Recommended Practice for Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," Institute of Electrical and Electronic Engineers.
10. TVA Calculation WBN EEB-MS-TI11-0003, "125 VDC Vital Battery and Charger Evaluation."
11. Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Subsystems," U.S. Nuclear Regulatory Commission, February 1978.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

manufacturer recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

The footnotes to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote b to Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge for vital batteries and < 1.0 amps for DG batteries. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 2). Footnote c to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 31 days following a battery recharge. Within 31 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 31 days.

REFERENCES

1. Watts Bar FSAR, Section 15, "Accident Analysis," and Section 6, "Engineered Safety Features."

2. IEEE-450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the AC, vital DC, and AC vital bus electrical power distribution subsystems are functioning properly, with all the required buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. Watts Bar FSAR, Section 8.0, "Electric Power," Section 15, "Accident Analysis," and Section 6, "Engineered Safety Features."

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.

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 moved to 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) System pumps.

The pumping action of the 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

(continued)

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.

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 the NRC Policy Statement.

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.

(continued)

BASES

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. A Note to the Applicability precludes use of LCO 3.0.4 to permit entry into MODE 6 while not meeting the boron concentration requirement.

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.

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)

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, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

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. Title 10, Code of Federal Regulations, Part 50, Appendix A, Section III, GDC 26, "Reactivity Control System Redundancy and Capability."
2. Watts Bar FSAR, Section 15, "Accident Analysis."

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 to the RCS.

For all other MODES, the boron dilution accident was analyzed and was found to be capable of being mitigated. A Note to the Applicability precludes use of LCO 3.0.4 to permit entry into MODE 6 while any unborated water source is not isolated.

ACTIONS

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

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the unit in compliance with this LCO. 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.

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

A.2

Preventing inadvertent dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water 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.

(continued)

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 Neutron Monitoring System (NMS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed primary source range neutron flux monitors are fission chambers. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) with an instrument accuracy of 5% of the countrate. The detectors also provide continuous visual indication in the control room and an audible alarm to alert operators to a possible dilution accident. The NMS 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."

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

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)

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.

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 temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a note that allows only one RHR loop to be OPERABLE and in operation prior to the initial criticality of the unit. The note also allows the loop to be removed from service for up to 1 hour per 8-hour period provided no operations are permitted that would cause a dilution of RCS boron concentration. This allowance is provided only for the initial criticality since there is no decay heat present.

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." A Note to the Applicability precludes use of LCO 3.0.4 to permit entry into MODE 6 with water level < 23 ft above the top of the reactor vessel flange.

(continued)

BASES

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, actions 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.

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

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.

(continued)

BASES

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 of the reactor vessel nozzles, the RHR pump suction 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.

REFERENCES

1. Watts Bar FSAR, Section 5.5.7, "Residual Heat Removal System."

B 3.9 REFUELING OPERATIONS

B 3.9.9 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND	The spent fuel storage rack criticality analysis assumes 2000 ppm soluble boron in the fuel pool during a dropped/misplaced fuel assembly event.
APPLICABLE SAFETY ANALYSES	This requirement ensures the presence of at least 2000 ppm soluble boron in the spent fuel pool water as assumed in the spent fuel rack criticality analysis for dropped/misplaced fuel assembly event. The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.
LCO	The LCO requires that the boron concentration in the spent fuel pool be greater than or equal to 2000 ppm during fuel movement.
APPLICABILITY	This LCO is applicable when the spent fuel pool is flooded and fuel is being moved. Once fuel movement begins, the movement will continue until the configuration of the assemblies in the storage racks is verified to comply with the criticality loading criteria specified in Specification 4.3.1.1.d.
ACTIONS	<u>A.1</u> If the spent fuel pool boron concentration does not meet the above requirements, fuel handling in the spent fuel pool must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis. Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.9.9.1</u> This SR requires that the spent fuel pool boron concentration be verified greater than or equal to 2000 ppm. This surveillance is to be performed prior to movement of fuel in the spent fuel pool and at least once ever 72 hours thereafter during the movement of fuel in the spent fuel pool. The Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of the sample. The Frequency is based on operating experience, which has shown 72 hours to be adequate.
REFERENCES	1. Watts Bar FSAR, Section 15, "Accident Analysis."

<u>LIST OF TABLES</u>

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
i	0	11/94
ii	0	11/94
iii	0	11/94
iv	0	11/94
v	0	05/95
vi	0	11/94
vii	0	11/94
viii	0	05/95
ix	0	05/95
x	0	05/95
xi	0	05/95
xii	0	05/95
xiii	0	05/95

PAGE	REVISION	DATE
1.1-1	0	11/94
1.1-2	0	11/94
1.1-3	0	11/94
1.1-4	0	11/94
1.1-5	0	11/94
1.1-6	0	11/94
1.2-1	0	11/94
1.2-2	0	11/94
1.2-3	0	11/94
1.3-1	0	11/94
1.3-2	0	11/94
1.3-3	0	11/94
1.3-4	0	11/94
1.3-5	0	11/94
1.3-6	0	11/94
1.3-7	0	11/94
1.3-8	0	11/94
1.3-9	0	11/94
1.3-10	0	11/94
1.3-11	0	11/94
1.3-12	0	11/94
1.3-13	0	11/94
1.4-1	0	11/94
1.4-2	0	11/94
1.4-3	0	11/94
1.4-4	0	11/94
3.0-1	0	11/94
3.0-2	0	05/95
3.0-3	0	11/94
3.0-4	0	05/95
3.1-1	0	11/94
3.1-2	0	11/94
3.1-3	0	11/94
3.1-4	0	11/94
3.1-5	0	11/94

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
3.1-6	0	11/94
3.1-7	0	11/94
3.1-8	0	11/94
3.1-9	0	11/94
3.1-10	0	11/94
3.1-11	0	11/94
3.1-12	0	11/94
3.1-12a	0	11/94
3.1-13	0	05/95
3.3-1	0	11/94
3.3-2	0	11/94
3.3-3	0	11/94
3.3-4	0	11/94
3.3-5	0	05/95
3.3-6	0	11/94
3.3-7	0	05/95
3.3-8	0	05/95
3.3-9	0	05/95
3.3-10	0	05/95
3.3-11	0	05/95
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3.4-4	0	11/94
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3.6-1	0	11/94
3.6-2	0	11/94
3.6-3	0	11/94
3.6-4	0	11/94
3.6-5	0	11/94
3.6-6	0	11/94
3.6-7	0	11/94
3.7-1	0	05/95
3.7-2	0	11/94

TECHNICAL REQUIREMENTS

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
3.7-3	0	05/95
3.7-4	0	05/94
3.7-5	0	11/94
3.7-6	0	11/94
3.7-7	0	11/94
3.7-8	0	11/94
3.7-9	0	11/94
3.7-10	0	11/94
3.7-11	0	11/94
3.7-12	0	11/94
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3.8-4	0	11/94
3.8-5	0	11/94
3.8-6	0	11/94
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3.8-17	0	11/94
3.8-18	0	11/94
3.9-1	0	11/94
3.9-2	0	11/94
3.9-3	0	11/94
3.9-4	0	11/94
3.9-5	0	11/94
5.0-1	0	11/94

TECHNICAL REQUIREMENTS BASES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.0-1	0	11/94
B 3.0-2	0	11/94
B 3.0-3	0	11/94
B 3.0-4	0	11/94
B 3.0-5	0	05/95
B 3.0-6	0	05/95
B 3.0-7	0	05/95
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B 3.0-9	0	11/94
B 3.0-10	0	11/94
B 3.0-11	0	11/94
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B 3.0-13	0	11/94
B 3.0-14	0	05/95
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B 3.1-10	0	11/94
B 3.1-11	0	11/94
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B 3.1-14	0	11/94
B 3.1-15	0	11/94
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B 3.1-17	0	11/94
B 3.1-18	0	11/94
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B 3.3-3	0	11/94
B 3.3-4	0	11/94
B 3.3-5	0	11/94
B 3.3-6	0	11/94
B 3.3-7	0	11/94
B 3.3-8	0	11/94
B 3.3-9	0	11/94
B 3.3-10	0	11/94

TECHNICAL REQUIREMENTS BASES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.3-11	0	11/94
B 3.3-12	0	11/94
B 3.3-13	0	11/94
B 3.3-14	0	11/94
B 3.3-15	0	11/94
B 3.3-16	0	11/94
B 3.3-17	0	11/94
B 3.3-18	0	11/94
B 3.3-19	0	11/94
B 3.3-20	0	11/94
B 3.4-1	0	11/94
B 3.4-2	0	11/94
B 3.4-3	0	11/94
B 3.4-4	0	11/94
B 3.4-5	0	11/94
B 3.4-6	0	11/94
B 3.4-7	0	11/94
B 3.4-8	0	11/94
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B 3.4-10	0	11/94
B 3.4-11	0	11/94
B 3.4-12	0	11/94
B 3.4-13	0	11/94
B 3.4-14	0	11/94
B 3.4-15	0	11/94
B 3.4-16	0	11/94
B 3.4-17	0	11/94
B 3.6-1	0	11/94
B 3.6-2	0	11/94
B 3.6-3	0	11/94
B 3.6-4	0	11/94
B 3.6-5	0	11/94
B 3.6-6	0	11/94
B 3.6-7	0	11/94
B 3.6-8	0	11/94
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B 3.6-10	0	11/94
B 3.6-11	0	11/94
B 3.6-12	0	11/94
B 3.7-1	0	05/95
B 3.7-2	0	05/95
B 3.7-3	0	11/94
B 3.7-4	0	05/95
B 3.7-5	0	11/94
B 3.7-6	0	11/94

LIST OF EFFECTIVE PAGES

3.0 TECHNICAL REQUIREMENT (TR) APPLICABILITY

TR 3.0.4 (continued)

specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

TR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

Exceptions to this Requirement are stated in the individual Requirements. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

TR 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to TR 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

TR 3.0.6

When a supported system TR or LCO is not met solely due to a support system TR or LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system TR or LCO ACTIONS are required to be entered. This is an exception to TR 3.0.2 and LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Technical Specification 5.7.2.18, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the TR or LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with TR 3.0.2 and LCO 3.0.2.

3.0 TECHNICAL SURVEILLANCE REQUIREMENT (TSR) APPLICABILITY

TSR 3.0.3 (continued)	When the Surveillance is performed within the delay period and the Surveillance is not met, the TR must immediately be declared not met, and the applicable Condition(s) must be entered.
TSR 3.0.4	Entry into a MODE or other specified condition in the Applicability of an TR shall not be made unless the TR's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. TSR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

TR 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.7 Position Indication System, Shutdown

TR 3.1.7 The group demand position indicators shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown or control rod that is not fully inserted.

APPLICABILITY: MODES 3, 4, and 5, when the reactor trip breakers are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more group demand position indicators inoperable.	A.1 Open reactor trip breakers.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.7.1	Determine that each group demand position indicator is OPERABLE by movement of the associated shutdown or control rod 10 steps in any one direction.	Within 4 hours after closing the reactor trip breakers AND 31 days thereafter

TR 3.3 INSTRUMENTATION

TR 3.3.2 Engineered Safety Features Actuation System Instrumentation

TR 3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks as shown in Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5, LCO 3.3.5; Technical Specification 3.3.6, Table 3.3.6-1; and Technical Specification 3.6.9 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3.2-1 of this document.

APPLICABILITY: As shown in Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5 Applicability; and Technical Specification 3.3.6 Applicability; and Technical Specification 3.6.9 Applicability.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refer to Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5 Actions; Technical Specification 3.3.6 Actions; and Technical Specification 3.6.9 Actions.	A.1 Refer to Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5 Actions; Technical Specification 3.3.6 Actions; and Technical Specification 3.6.9 Actions.	Refer to Technical Specification 3.3.2, Table 3.3.2-1; Technical Specification 3.3.5 Actions; Technical Specification 3.3.6 Actions; and Technical Specification 3.6.9 Actions.

Table 3.3.2-1 (Page 1 of 5)

Engineered Safety Features Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Ventilation Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Raw Cooling Water	N.A.
j. CREVS Actuation	N.A.
k. Containment Air Return Fan	N.A.
l. Component Cooling System	N.A.
m. Start Diesel Generators	N.A.
n. Reactor Trip	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 27(4)/32(14)/37(5)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8(3)
3) Containment Isolation-Phase "A"(6)	≤ 12(2)/22(1)
4) Containment Ventilation Isolation	≤ 5.0(2)(11)
5) Auxiliary Feedwater Pumps	≤ 60(10)
6) Essential Raw Cooling Water	≤ 47(2)/57(1)
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50(2)/60(1)
9) Start Diesel Generators	≤ 12(12)
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27(4)/32(14)/37(5)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8(3)
3) Containment Isolation-Phase "A"(6)	≤ 12(2)/22(1)
4) Containment Ventilation Isolation	≤ 5.0(2)(11)
5) Auxiliary Feedwater Pumps	≤ 60(10)

(continued)

Table 3.3.2-1 (Page 2 of 5)

Engineered Safety Features Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low (continued)	
6) Essential Raw Cooling Water	≤ 47(2)/57(1)
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50(2)/60(1)
9) Start Diesel Generators	≤ 12(12)
4. Steam Line Pressure Negative Rate-High	
a. Steam Line Isolation	≤ 7
5. Steam Line Pressure - Low	
a. Safety Injection (ECCS)	≤ 27(4)/32(14)/37(5)
1) Reactor Trip (from SI)	≤ 2
2) Feedwater Isolation	≤ 8(3)
3) Containment Isolation-Phase "A"(6)	≤ 12(2)/22(1)
4) Containment Ventilation Isolation	≤ 5.0(2)(11)
5) Auxiliary Feedwater Pumps	≤ 60(10)
6) Essential Raw Cooling Water	≤ 47(2)/57(1)
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50(2)/60(1)
9) Start Diesel Generators	≤ 12(12)
b. Steam Line Isolation	≤ 7
6. Containment Pressure - High - High	
a. Containment Spray	≤ 221(13)
b. Containment Isolation-Phase "B"	≤ 68(2)/78(1)
c. Steam Line Isolation	≤ 7
d. Containment Air Return Fans	480 ≤ RT ≤ 600
7. Steam Generator Water Level - High - High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 8(3)

(continued)

Table 3.3.2-1 (Page 3 of 5)

Engineered Safety Features Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Steam Generator Water Level - Low - Low Coincident with Vessel $\Delta T \leq$ 50% RTP	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq 60^{(7)}$
b. Turbine-driven Auxiliary Feedwater Pumps	$\leq 60^{(8)}$
9. Steam Generator Water Level-Low-Low Coincident with Vessel ΔT > 50% RTP	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq 60^{(7)}$
b. Turbine-driven Auxiliary Feedwater Pumps	$\leq 60^{(8)}$
10. RWST Level-Low Coincident with Containment Sump Level-High and Safety Injection	
Automatic Switchover to Containment Sump	≤ 250
11. Loss-of-Offsite Power	
Auxiliary Feedwater Pumps	≤ 60
12. Trip of All Main Feedwater Pumps	
Auxiliary Feedwater Pumps	≤ 60
13. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 40
b. Turbine-driven Auxiliary Feedwater Pumps	≤ 38
14. Loss of Voltage/Degraded Voltage	
6.9 kV Shutdown Board	$\leq 12^{(9)}$
15. MSV Vault Room Water Level - High	
a. North MSV Vault Room	$\leq 8.5^{(15)}/12^{(16)}$
b. South MSV Vault Room	$\leq 8.5^{(15)}/12^{(16)}$

Table 3.3.2-1 (Page 4 of 5)

Engineered Safety Features Response Times

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air operated valves.

(4) Offsite power available - diesel generator starting and sequence loading delays not included. Response time limit includes the opening of valves to establish flowpath and bringing the pumps to full speed. The additional sequential transfer of CCP suction from the VCT to the RWST (RWST valves open, then the VCT valves close) is included.

(5) Diesel generator starting and sequence loading delays included. Response time limit includes the opening of valves to establish flow path and bringing the pumps up to full speed. The additional sequential transfer of suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(6) The following equipment are exceptions to the response time shown in the table and will have the following response times for the initiating signals and functions:

A. Fire Protection CIVs	22(2)/32(1)
B. Ice Condenser CIVs	32
C. Excess Letdown Hx Supply CIV	68(2)/78(1)
D. EGTS Fans	20(2)/30(1)
E. Required for EGTS OPERABILITY	
1. Fire Protection Secondary CIVs	20(2)/30(1)
2. Secondary Containment Purge Isolation Valves	12.7(2)/22.7(1)
F. Steam Generator Blowdown CIVs	17(2)/27(1)

(7) On 2/3 any steam generator and Trip Time Delay = 0 seconds.

(8) On 2/3 in 2/4 steam generators and Trip Time Delay = 0 seconds.

(9) The response time is measured from the time the 6.9 kV shutdown boards voltage exceeds the Setpoint until the time full voltage is returned for the loss of voltage sensors; or from the time the degraded voltage timers generate a signal to trip the feeder breakers and shed loads until the time full voltage is returned for the degraded voltage sensors.

(10) The Response Time for motor-driven AFW pumps includes the diesel generator starting and sequence loading delays. The Response Time for (steam) turbine driven AFW pumps does not include diesel generator starting and sequence loading delays.

(continued)

Table 3.3.2-1 (Page 5 of 5)

Engineered Safety Features Response Times

TABLE NOTATIONS

(11) Containment purge valves only. Containment radiation monitor valves have a response time of 6.5 seconds.

(12) Diesel generator start time includes a reactor trip response time of 2 seconds.

(13) Includes diesel generator starting, containment spray pump sequence loading-delay/breaker closure, plus stroke time of 1-FCV-72-39/2.

(14) Diesel generator starting and sequence loading delays included. Response time limit includes the opening of valves to establish flowpath and bring pumps to full speed. The additional sequential transfer of ECCS pump suction from the VCT to the RWST (RWST valves open) is included.

(15) Feedwater Isolation Valve (motor) and Feedwater Regulating Valve (air operated) response time includes an ESFAS signal response time of 2 seconds.

(16) Feedwater pumps coast down to zero flow response time includes an ESFAS signal response time of 2 seconds.

TR 3.4 REACTOR COOLANT SYSTEM (RCS)

TR 3.4.3 Reactor Coolant System Vents

TR 3.4.3 Two Reactor Coolant System Vent (RCSV) paths shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCSV path inoperable.	A.1 Initiate action to maintain the affected RCSV path closed with power removed from the valve actuators. AND A.2 Restore the inoperable path to OPERABLE status.	Immediately 30 days
B. Two RCSV paths inoperable.	B.1 Restore one RCSV path to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.3.1	Verify that the upstream manual RCSV isolation valve is locked in the opened position.	18 months
TSR 3.4.3.2	Operate each remotely controlled valve through at least one complete cycle of full travel from the control room.	In accordance with the Inservice Testing Program
TSR 3.4.3.3	Verify flow through the RCSV paths during venting.	18 months

TR 3.7 PLANT SYSTEMS

TR 3.7.1 Steam Generator Pressure/Temperature Limitations

TR 3.7.1 The pressure of the reactor and secondary coolants in the Steam Generators shall be ≤ 200 psig.

APPLICABILITY: Whenever the temperature of the reactor or secondary coolant in any Steam Generator ≤ 70°F.

-----------------------------NOTE-----------------------------
While this TR is not met, decreasing the temperature of the coolant in the primary or secondary of any steam generator to ≤ 70°F is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------NOTE--------- All Required Actions must be completed whenever this Condition is entered. --------------------		
Steam Generator pressure not within limits.	A.1 Reduce pressure to ≤ 200 psig.	30 minutes
	AND	
	A.2 Perform an engineering evaluation to determine the effect of the over-pressurization on the structural integrity of the Steam Generator.	Prior to increasing Steam Generator coolant temperatures to > 200°F.
	AND	
		(continued)

TR 3.7 PLANT SYSTEMS

TR 3.7.2 Flood Protection Plan

TR 3.7.2 The flood protection plan shall be ready for implementation to maintain the plant in a safe condition.

APPLICABILITY: When one or more of the following conditions exist:

a. Flood-producing rainfall conditions in the east Tennessee watershed, or

b. An early warning or alert that a critical combination of flood and/or high headwater levels may or have developed, or

c. An early warning or alert involving Fontana Dam, or

d. Recognizable seismic activity in the east Tennessee region.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Stage I flood warning issued.	A.1 Be in at least MODE 3.	6 hours
	<u>AND</u>	
	A.2 Initiate and complete the Stage I flood protection plan.	10 hours
	<u>AND</u>	
	A.3 Establish a SDM $\geq$ 5% $\Delta k/k$ and $T_{avg} \leq$ 350°F.	10 hours
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4.1 Verify that communications between the TVA Division of Water Resources and the Watts Bar Nuclear Plant have been established. OR A.4.2 Initiate and complete the Stage II flood protection procedure.	10 hours 27 hours
B. Stage II flood warning issued.	B.1 Initiate and complete the Stage II flood protection plan.	17 hours OR Prior to predicted flooding of the site.
C. Seismic event occurs after a critical combination of flood and/or headwater alerts are issued.	C.1 Verify and maintain communications between TVA Power Control Center and the Watts Bar Nuclear Plant. OR	6 hours (continued)

Table 3.7.3-4 (Page 1 of 2)

Functional Testing General Notes

1. The representative sample of snubbers shall include each type and shall be tested using sample plan A for hydraulic snubbers and sample plan B for mechanical snubbers.

2. The NRC Regional Administrator shall be notified in writing of any changes to the sample plan prior to the test period.

SAMPLE PLAN A

1. At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test.

2. For each snubber of a type that does not meet the functional test acceptance criteria of Table 3.7.3-5, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

SAMPLE PLAN B

1. An initial representative sample of 37 snubbers shall be functionally tested in accordance with Figure 3.7.3-1. For each snubber type which does not meet the functional test acceptance criteria of Table 3.7.3-5, another sample of at 19 snubbers shall be tested. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 36.49 + 18.18C. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

Table 3.7.3-4 (Page 2 of 2)

Functional Testing General Notes

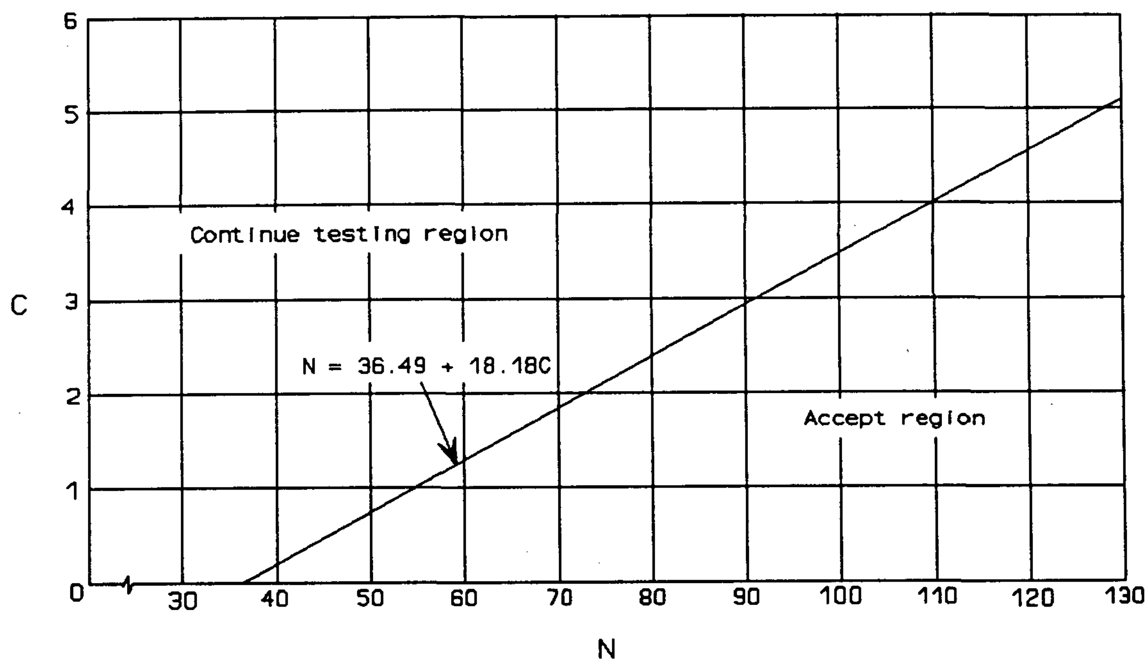
TABLE NOTES

1. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested.

2. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type.

3. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan.

4. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.



N = Total number of snubbers tested

C = Number of snubbers which do not meet functional test acceptance criteria

FIGURE 3.7.3-1

Sample Plan B for Snubber Functional Test

TR 3.7 PLANT SYSTEMS

TR 3.7.5 Area Temperature Monitoring

TR 3.7.5 The normal temperature limit of each area shown in Table 3.7.5-1 shall not be exceeded for > 8 hours and the abnormal temperature limits shall not be exceeded.

APPLICABILITY: Whenever the affected equipment in an area is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more areas exceeding normal temperature limits for > 8 hours.	------------NOTE------------ TR 3.0.3 and TR 3.0.4 are not applicable. ---------------------------- A.1 Prepare and submit to the NRC a report in accordance with 10 CFR 50.4 that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	30 days (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more areas exceeding abnormal temperature limits except for the IPS Mechanical or Electrical Equipment Rooms (Areas 31, 32, or 34). OR Mechanical or Electrical Equipment Rooms (Areas 23, 25, 35, or 36) temperature less than 35°F.	B.1.1 Restore the area(s) to within normal temperature limits. OR B.1.2 Declare the affected equipment in the affected area(s) inoperable. AND B.2 Prepare and submit to the NRC a report in accordance with 10 CFR 50.4 that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	4 hours 4 hours 30 days
C. Mechanical or Electrical Equipment Rooms in Intake Pumping Station (Areas 31, 32, or 34) less than 40 °F and greater than 32 °F.	C.1 Initiate action to maintain temperatures greater than 32 °F. AND C.2 Restore temperatures to within normal limits.	24 hours 7 days

D.	Mechanical or Electrical Equipment Rooms in Intake Pumping Station (Areas 31, 32, or 34) 32 °F or less.	D.1	Declare the affected equipment in the affected area(s) inoperable.	Immediately
		<u>AND</u>		
		D.2	Prepare and submit to the NRC a report in accordance with 10 CFR 50.4 that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	30 days

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.5.1	Verify each area temperature is within limits.	12 hours

Table 3.7.5-1
Area Temperature Monitoring

AREA	NORMAL LIMIT °F	ABNORMAL LIMIT °F
1. Aux Bldg el 772 next to 480V Sd Bd transformer 1A2-A.	≤ 104	≤ 110
2. Aux Bldg el 772 next to 480V Sd Bd transformer 1B1-B.	≤ 104	≤ 110
3. Aux Bldg el 772 next to 480V Sd Bd transformer 2A2-A.	≤ 104	≤ 110
4. Aux Bldg el 772 next to 480V Sd Bd transformer 2B2-B.	≤ 104	≤ 110
5. Aux Bldg el 772 next to 480V Rx MOV Bd 1A2-A.	≤ 83	≤ 104
6. Aux Bldg el 772 next to 480V Rx MOV Bd 2A2-A.	≤ 83	≤ 104
7. Aux Bldg el 772 next to 480V Rx MOV Bd 2B2-B.	≤ 83	≤ 104
8. Aux Bldg el 772 across from spare 125V vital battery charger 1-S.	≤ 83	≤ 104
9. Aux Bldg el 772 U1 Mech Equip Room.	≤ 91	≤ 104
10. Aux Bldg el 757 U1 Sd Bd room behind stairs S-A3.	≤ 85	≤ 104
11. Aux Bldg el 757 U2 Sd Bd room behind stairs S-A13.	≤ 85	≤ 104
12. Aux Bldg el 757 U1 Refueling beside Aux boration makeup tk.	≤ 104	≤ 115
13. Aux Bldg el 737 U1 outside supply fan room.	≤ 104	≤ 110
14. Aux Bldg el 713 U1 across from AFW pumps.	≤ 104	≤ 110
15. Aux Bldg el 692 U1 outside AFW pump room door.	≤ 104	≤ 110
16. Aux Bldg el 692 U2 near boric acid concentrate filter vault.	≤ 104	≤ 110
17. Aux Bldg el 676 next to 0-L-629.	≤ 104	≤ 110
18. North steam valve vault room U1. (at MSSVs)	≥ 80	≥ 80

(continued)

Table 3.7.5-1
Area Temperature Monitoring

AREA	NORMAL LIMIT °F	ABNORMAL LIMIT °F
19. South steam valve vault room U1. (at MSSVs)	≥ 80	≥ 80
20. Add Equip Bldg U1 el 729 between UHI accumulators.	≤ 92	≤ 110
21. CB Main Control Room south wall.	≤ 80	≤ 104
22. CB Main Control Room across from 1-M-9.	≤ 80	≤ 104
23. CB Computer room el 708 center of room.	≤ 74	≤ 104
24. CB Aux. Instrument Room el 708.	≤ 90	≤ 104
25. D/G Bldg el 742 2B-B D/G room on wall by battery charger.	≤ 104	≤ 120
26. D/G Bldg el 742 1A-A D/G Room near D/G set.	≥ 50	≥ 50
27. D/G Bldg el 742 1B-B D/G Room near D/G set.	≥ 50	≥ 50
28. D/G Bldg el 742 2A-A D/G Room near D/G set.	≥ 50	≥ 50
29. D/G Bldg el 742 2B-B D/G Room near D/G set.	≥ 50	≥ 50
30. D/G Bldg el 760.5 next to 480V diesel Aux Bd 2B1-B.	≤ 104	≤ 120
31. IPS Mechanical Equipment Room 1 el 722 near ERCW and HPFP Instruments and sense lines.	≥50≤104	≥40≤115
32. IPS Mechanical Equipment Room 2 el 722 near ERCW and HPFP Instruments and sense lines.	≥50≤104	≥40≤115
33. IPS el 741 in B train ERCW pump room.	≤ 120	≤ 120
34. IPS el 711 next to 480V IPS board and transformer (A bus).	≥50≤104	≥40≤115
35. IPS el 711 next to 480V IPS board and transformer (B bus).	≤ 104	≤ 115
36. Add D/G Bldg el 742 C-S D/G Room near D/G set.	≥ 50	≥ 50

Table 3.8.3-1 (Page 1 of 5)

Motor-Operated Valves Thermal Overload
Devices Which Are Bypassed Under
Accident Conditions

VALVE NO.	FUNCTION
1-FCV-62-63	Isolation for Seal Water Filter
1-FCV-62-90	ECCS Operation
1-FCV-62-91	ECCS Operation
1-FCV-62-61	Cont. Isolation
1-LCV-62-132	ECCS Operation
1-LCV-62-133	ECCS Operation
1-LCV-62-135	ECCS Operation
1-LCV-62-136	ECCS Operation
1-FCV-74-1	Open for Normal Plant Cooldown
1-FCV-74-2	Open for Normal Plant Cooldown
1-FCV-74-3	ECCS Operation
1-FCV-74-21	ECCS Operation
1-FCV-74-12	RHR Pump, Mini-flow Protects Pump
1-FCV-74-24	RHR Pump, Mini-flow Protects Pump
1-FCV-74-33	ECCS Operation
1-FCV-74-35	ECCS Operation
1-FCV-63-7	ECCS Operation
1-FCV-63-6	ECCS Operation
1-FCV-63-156	ECCS Flow Path
1-FCV-63-157	ECCS Flow Path
1-FCV-63-25	BIT Injection
1-FCV-63-26	BIT Injection
1-FCV-63-1	ECCS Operation
1-FCV-63-72	ECCS Flow Path from Cont. Sump
1-FCV-63-73	ECCS Flow Path from Cont. Sump
1-FCV-63-8	ECCS Flow Path
1-FCV-63-11	ECCS Flow Path
1-FCV-63-93	ECCS Cooldown Flow Path
1-FCV-63-94	ECCS Cooldown Flow Path

(continued)

Table 3.8.3-1 (Page 2 of 5)

Motor-Operated Valves Thermal Overload Devices Which Are Bypassed Under Accident Conditions

(continued)

VALVE NO.	FUNCTION
1-FCV-63-172	ECCS Flow Path
1-FCV-63-5	ECCS Flow Path
1-FCV-63-47	Train Isolation
1-FCV-63-48	Train Isolation
1-FCV-63-4	SI Pump Mini-flow
1-FCV-63-175	SI Pump Mini-flow
1-FCV-63-3	SI Pump Mini-flow
1-FCV-63-152	ECCS Recirc
1-FCV-63-153	ECCS Recirc
1-FCV-3-33	Quick Closing Isolation
1-FCV-3-47	Quick Closing Isolation
1-FCV-3-87	Quick Closing Isolation
1-FCV-3-100	Quick Closing Isolation
1-FCV-1-15	Steam Supply to Aux FWP Turbine
1-FCV-1-16	Steam Supply to Aux FWP Turbine
1-FCV-3-179A	ERCW System Supply to Pump
1-FCV-3-179B	ERCW System Supply to Pump
1-FCV-3-136A	ERCW System Supply to Pump
1-FCV-3-136B	ERCW System Supply to Pump
1-FCV-3-116A	ERCW System Supply to Pump
1-FCV-3-116B	ERCW System Supply to Pump
1-FCV-3-126A	ERCW System Supply to Pump
1-FCV-3-126B	ERCW System Supply to Pump
1-FCV-70-133	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-143	Isolation for Excess Letdown Ht Xchngr
1-FCV-70-92	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-90	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-87	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-89	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-140	Isolation for RCP Oil Coolers & Therm B
1-FCV-70-134	Isolation for RCP Oil Coolers & Therm B

(continued)

Table 3.8.3-1 (Page 3 of 5)

Motor-Operated Valves Thermal Overload
Devices Which Are Bypassed Under
Accident Conditions

(continued)

VALVE NO.	FUNCTION
1-FCV-67-123	CS Heat Exchanger Supply
1-FCV-67-125	CS Heat Exchanger Supply
1-FCV-67-124	CS Heat Exchanger Discharge
1-FCV-67-126	CS Heat Exchanger Discharge
1-FCV-67-146	CCWS Heat Exchanger Throttling
1-FCV-67-83	Containment Isolation Lower
1-FCV-67-88	Containment Isolation Lower
1-FCV-67-87	Containment Isolation Lower
1-FCV-1-51	AFPT Trip and Throttle Valve
1-FCV-67-95	Containment Isolation Lower
1-FCV-67-96	Containment Isolation Lower
1-FCV-67-91	Containment Isolation Lower
1-FCV-67-103	Containment Isolation Lower
1-FCV-67-104	Containment Isolation Lower
1-FCV-67-99	Containment Isolation Lower
1-FCV-67-111	Containment Isolation Lower
1-FCV-67-112	Containment Isolation Lower
1-FCV-67-107	Containment Isolation Lower
1-FCV-67-130	Containment Isolation Upper
1-FCV-67-131	Containment Isolation Upper
1-FCV-67-295	Containment Isolation Upper
1-FCV-67-134	Containment Isolation Upper
1-FCV-67-296	Containment Isolation Upper
1-FCV-67-133	Containment Isolation Upper
1-FCV-67-139	Containment Isolation Upper
1-FCV-67-297	Containment Isolation Upper
1-FCV-67-138	Containment Isolation Upper
1-FCV-67-142	Containment Isolation Upper
1-FCV-67-298	Containment Isolation Upper

(continued)

Table 3.8.3-1 (Page 4 of 5)

Motor-Operated Valves Thermal Overload Devices Which Are Bypassed Under Accident Conditions

(continued)

VALVE NO.	FUNCTION
1-FCV-67-141	Containment Isolation Upper
1-FCV-72-21	Cont. Spray Pump Suction
1-FCV-72-22	Cont. Spray Pump Suction
1-FCV-72-2	Cont. Spray Isolation
1-FCV-72-39	Cont. Spray Isolation
1-FCV-72-40	RHR Cont. Spray Isolation
1-FCV-72-41	RHR Cont. Spray Isolation
1-FCV-72-44	Cont. Sump to Hdr A - Cont. Spray
1-FCV-72-45	Cont. Sump to Hdr B - Cont. Spray
1-FCV-26-240	Containment Isolation
1-FCV-26-241	Annulus Isolation
1-FCV-26-242	Annulus Isolation
1-FCV-26-243	RCP Cont. Spray Isolation
1-FCV-26-244	Annulus Isolation
1-FCV-26-245	Annulus Isolation
1-FCV-68-332	RCS PRZR Rel.
1-FCV-68-333	RCS PRZR Rel.
1-FCV-70-153	RHR Ht Ex B-B Outlet
1-FCV-70-156	RHR Ht Ex A-A Outlet
1-FCV-70-207	Cont. Demin. Waste Evap. Bldg Supply
1-FCV-67-9A	ERCW Strainer Backwash
2-FCV-67-9A	ERCW Strainer Backwash
1-FCV-67-9B	ERCW Strainer Flush
2-FCV-67-9B	ERCW Strainer Flush
1-FCV-67-10A	ERCW Strainer Backwash
2-FCV-67-10A	ERCW Strainer Backwash
1-FCV-67-10B	ERCW Strainer Flush
2-FCV-67-10B	ERCW Strainer Flush

(continued)

Table 3.8.3-1 (Page 5 of 5)

Motor-Operated Valves Thermal Overload
Devices Which Are Bypassed Under
Accident Conditions

(continued)

VALVE NO.	FUNCTION
1-FCV-67-89	Containment Isolation
1-FCV-67-97	Containment Isolation
1-FCV-67-105	Lower Containment Isolation
1-FCV-67-113	Lower Containment Isolation
1-FCV-67-143	CCS Heat Exchanger Discharge
0-FCV-67-144	CCS Heat Exchanger Bypass
0-FCV-67-152	CCS Heat Exchanger Discharge
0-FCV-67-205	Nonessential Equipment Isolation
0-FCV-67-208	Station Service/Contr. Air Supply
1-FCV-70-183	Sample Ht Ex Header Outlet
1-FCV-70-100	RCP Oil Cooler Supply Cont. Isolation
0-FCV-70-197	SFPCS Ht Ex Supply Header
1-FCV-70-215	Sample Ht Ex Header Inlet
1-FCV-74-8	RHR Isolation Bypass
1-FCV-74-9	RHR Isolation Bypass

BASES

TRs

TR 3.0.3 (continued)

limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, TR 3.0.3 provides actions for Conditions not covered in other Requirements. The requirements of TR 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by TR 3.0.3. The requirements of TR 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Requirements sufficiently define the remedial measures to be taken.

Exceptions to TR 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with TR 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in TR 3.3.4, "Seismic Instrumentation". TR 3.3.4 has an Applicability of "At all times". Therefore, this TR can be applicable in any or all MODES. If the TR and the Required Actions of TR 3.3.4 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Actions are the appropriate Required Actions to complete in lieu of the actions of TR 3.0.3. These exceptions are addressed in the individual Requirements.

TR 3.0.4

TR 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when a TR is not met. It precludes placing the unit in a MODE or other

(continued)

BASES

TRs

TR 3.0.4 (continued)

specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the TR would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the TR requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Requirement should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of TR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of TR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to TR 3.0.4 are stated in the individual Requirements. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Requirement.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by TSR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with TR 3.0.4 or where an exception

(continued)

BASES

TRs

TR 3.0.4 (continued)

to TR 3.0.4 is stated, is not a violation of TSR 3.0.1 or TSR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, TSRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected TR.

TR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, TR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of TR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Requirements sufficiently define the remedial measures to be taken.

TR 3.0.5

TR 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Requirement is to provide an exception to TR 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of TSRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed TSRs. This requirement does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the TSRs.

(continued)

BASES

TRs

TR 3.0.5 (continued)

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of a TSR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of a TSR on another channel in the same trip system.

TR 3.0.6

TR 3.0.6 establishes an exception to TR 3.0.2 for support systems that have an TR specified in the Technical Requirements. This exception is provided because TR 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system TR be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system TR's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is a TR specified for it in the Technical Requirements, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' TRs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some

(continued)

BASES

TSRs

TSR 3.0.4

TSR 3.0.4 establishes the requirement that all applicable TSRs must be met before entry into a MODE or other specified condition in the Applicability.

This Requirement ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. However, in certain circumstances failing to meet an TSR will not result in TSR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated TSR(s) are not required to be performed per TSR 3.0.1, which state that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, TSR 3.0.4 does not apply to the associated TSR(s) since the requirement for the TSR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an TSR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the TR is not met in this instance, TR 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of this Requirement should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

(continued)

BASES

TSRs

TSR 3.0.4 (continued)

The provisions of TSR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of TR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of TSRs are specified such that exceptions to TSR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the TSRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated TR prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the TR Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of TSR's annotation is found in Section 1.4, Frequency.

TSR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, TSR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, 3, or 4. The requirements of TSR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Requirements sufficiently define the remedial measures to be taken.

BASES

BACKGROUND (continued)

3. Fuel Burnup - boron concentration must be decreased to compensate for fuel burnup and the buildup of fission products in the fuel.

4. Cold Shutdown - boron concentration must be increased to the cold shutdown concentration.

Boric acid is stored in three boric acid tanks. Two boric acid transfer pumps are provided for each unit with one pump normally aligned with one boric acid tank and continuously running at low speed to provide recirculation for the boric acid system and the boric acid tank. On a demand signal by the reactor makeup control system, the boric acid transfer pumps are shifted to high speed and the pump aligned to the makeup system delivers boric acid to the suction header of the charging pumps (Ref. 1).

APPLICABLE SAFETY ANALYSES

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. In the case of a malfunction of the CVCS, which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 2). OPERABILITY of the charging pumps, the RWST, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.

TR

TR 3.1.1 requires at least one boron injection flow path to be OPERABLE and capable of being powered from an OPERABLE emergency power source during MODES 4, 5, and 6 in order to provide a path to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations. This requirement may be achieved by meeting one of the following two conditions:

a. A flow path from an OPERABLE boric acid storage tank, through the boric acid transfer pump, through a charging pump to the RCS, or

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

acid storage tanks is required OPERABLE. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

TSR 3.1.1.2

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

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

REFERENCES

1. Watts Bar FSAR, Section 9.3.4, "Chemical and Volume Control System."
2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.
3. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar Nuclear Plant, Unit 1, Revision 00, April 1993.

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.2.1

This surveillance verifies the temperature of the required flow path from the boric acid tanks to be at least 63°F. This ensures that the high concentration of boric acid in the storage tanks is not allowed to precipitate due to cooling.

The surveillance is modified by a note stating that the surveillance is required only if the flow path from the boric acid storage tanks is used as one of the two required flow paths. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

TSR 3.1.2.2

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

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

TSR 3.1.2.3

This surveillance demonstrates that each automatic valve in the flow path actuates to its required position on an actual or simulated actuation signal. The 18-month Frequency was developed considering it is prudent that this surveillance only be performed during a plant outage. This is due to the plant conditions needed to perform the TSR and the potential for unplanned plant transients if the TSR is performed with the reactor at power.

(continued)

BASES (continued)

ACTIONS

A.1

With one or more group demand position indicators inoperable, the plant must be placed in a condition where the demand position indicators are not required. This is accomplished by opening the reactor trip breakers immediately.

The immediate Completion Time is consistent with the required time for actions to be pursued without delay and in a controlled manner.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.7.1

Exercising rods at a Frequency of 31 days allows the operator to determine that all withdrawn rods, including the group step counter demand position indicator, continue to be OPERABLE. A movement of 10 steps is adequate to demonstrate motion and verify a corresponding step change in the group step counter demand position indicator. Four hours is provided to perform the first surveillance after closing the reactor trip breakers. The 31-day Frequency takes into consideration other information available to the operator in the control room and the remote likelihood that rods would be withdrawn from fully inserted for extended periods of time during shutdown conditions.

REFERENCES

1. Watts Bar FSAR, Section 4.2.3 "Reactivity Control System."

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

BASES

APPLICABLE SAFETY ANALYSES (continued)

trip functions. RTS trip functions not specifically credited in the accident analyses have an N.A. response time requirement in Table 3.3.1-1. They are qualitatively credited in the safety analyses and the NRC staff-approved licensing basis for the plant. These RTS trip functions may provide protection for conditions which do not require dynamic transient analysis to demonstrate function performance. These RTS trip functions may also serve as backups to RTS trip functions that were credited in the accident analysis.

The safety analyses applicable to each RTS function are discussed in the bases for the Technical Specifications, B.3.3.1 (Ref. 4).

TR

OPERABILITY requirements for the RTS Instrumentation and interlocks are specified in Technical Specifications, section 3.3.1. TR 3.3.1 requires the RTS Instrumentation and interlocks of Table 3.3.1-1 of the TR to be OPERABLE with RESPONSE TIMES as shown in the table. RESPONSE TIMES must be within the specified limits for the affected instruments to be considered OPERABLE.

APPLICABILITY

Applicable MODES for the specific RTS Instrumentation and interlocks are delineated in Table 3.3.1-1 of Reference 4. The bases for Applicability of each function is included in Reference 4.

ACTIONS

A.1

The Required Actions for inoperable instruments are found in Reference 4. With one or more RESPONSE TIMES outside the specified limits, the affected instrument(s) must be considered inoperable and the appropriate Action referenced in Table 3.3.1-1 of Reference 4 must be taken. The bases for these actions is found in Reference 4.

(continued)

BASES

ACTIONS

B.1 (continued)

and correcting problems which could be associated with an inoperable path.

C.1 AND C.2

If the Required Action and associated Completion Time of Condition A or B are not met, the plant must be placed in a condition in which the TR does not apply. This is accomplished by placing the unit in MODE 3 within 6 hours and MODE 5 in an additional 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.3.1

Every 92 days it is necessary to verify the function of the block valve to isolate a failed open RCS vent valve. Cycling the block valve closed and open demonstrates its capability to perform this function. These valves are required to seat against full operational pressure with the down stream side at the pressure of the Pressurizer Relief Tank. The valves are solenoid-to-open and spring-to-close valves.

TSR 3.4.3.1, TSR 3.4.3.2 and TSR 3.4.3.3

Every 18 months it is necessary to verify that each of the two vent paths are OPERABLE. This verification consists of checking the upstream isolation valve and ensuring that the valve is locked in the open position. Further, the two control valves are operated from the control room, in accordance with the Inservice Testing Program through one complete cycle of full travel. Lastly, the test includes a verification of flow through the two vent paths.

REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements".

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.7 PLANT SYSTEMS

B 3.7.1 Steam Generator Pressure/Temperature Limitations

BASES

BACKGROUND	In order to meet regulatory and code requirements with respect to material toughness, certain limits on steam generator pressure and temperature are established. Material toughness varies with temperature and is lower at room temperature than at operating temperature. One indicator of the temperature effect on ductility is the nil-ductility temperature (NDT). Therefore, a nil-ductility reference temperature (RT_{NDT}) has been determined by experimental means. The RT_{NDT} is that temperature below which brittle (non-ductile) fracture may occur. For the steam generator, the RT_{NDT} has been determined to be 10°F for steam generators A, C, and D and 30°F for steam generator B (Ref. 1). Considering uncertainties and proper margins, the minimum operating temperature has been determined to be 70°F. The 70°F temperature must be established before the pressure is increased to 200 psig. This limitation on steam generator pressure and temperature, ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The fracture mechanic methodology, which is used to determine the stresses and material toughness, follows the guidance given by 10 CFR 50, Appendix G (Ref. 2). Reference 1 mandates the use of ASME Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3).
APPLICABLE SAFETY ANALYSIS	The RT_{NDT} limit is not derived from the Design Basis Accident analyses. The RT_{NDT} limit is imposed during normal operation to avert encountering pressure/temperature combinations which are not analyzed as part of the steam generator design. Unanalyzed pressure/temperature combinations could cause propagation of minor, undetected flaws, which could cause brittle failure of the pressure boundary. Because the RT_{NDT} limit is related to normal operation, the RT_{NDT} limit is not a consideration in designing the accident sequences for theoretical hazard evaluations (Ref. 4).

(continued)

BASES (continued)

TR

TR 3.7.1 requires that the pressures on the primary and the secondary sides in the steam generator are kept at or below 200 psig when the temperature is less than or equal to 70°F. The pressure induced stress from the 200 psig pressure is low enough to be insignificant, even at temperatures at or below RT_{NDT}.

APPLICABILITY

The operating requirements which must be observed to avoid a condition, which could lead to brittle failure, is not strictly limited to specific MODES. Hence, in general, Applicability should be at all times. However, in practice it is unlikely that these limits will be violated in the lower numbered MODES, due to the high operating temperature on the primary as well as the secondary side in the steam generators. Accordingly, the limits are most easily violated at low temperature, during shutdown and startup of the plant. Applicability can therefore conveniently be limited to whenever the temperature on the primary or the secondary side is at or below 70°F. A Note to the Applicability precludes use of TR 3.0.4 to permit decreasing the temperature of the coolant in the primary or secondary of any steam generator to ≤ 70°F while the pressure is > 200 psig.

ACTIONS

<u>A.1, A.2, and A.3</u>

With the combination of pressure and temperature not within limits, a reduction in pressure at or below 200 psig is required within 30 minutes. An engineering evaluation must be performed to determine the effect on the structural integrity of the pressure boundary. The evaluation must be finished and the conclusion made that no hazard exists, before the temperature is increased to more than 200°F. Condition A is modified by a Note which states that whenever Condition A is entered, all ACTIONS A.1 through A.3 must be completed.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.2 Flood Protection Plan

BASES

BACKGROUND

Nuclear power plants are designed to prevent the loss of capability for cold shutdown and maintenance thereof resulting from the most severe flood conditions that can reasonably be predicted to occur at the site as a result of severe hydrometeorological conditions, seismic activity, or both (Ref. 1). Assurance that safety-related facilities are capable of surviving all possible flood conditions is provided by the flood protection plan.

The elevations of plant features which could be affected by the submergence during floods vary from 714.5 ft Mean Sea Level (MSL) (access to electrical conduits) to 740.1 ft MSL (including wave runup). Plant grade is elevation 728 ft MSL which can be exceeded by rainfall floods and by seismic-caused dam failure floods. One kind of warning plan is needed to assure plant safety from rainfall floods, and another kind of warning plan is needed for seismic-caused dam failure floods.

The warning plan is divided into two stages. This two-stage scheme is designed to prevent excessive economic loss in case a potential flood does not fully develop. Stage I, which is a minimum of 10 hours, allows preparation steps, causing some damage to be sustained, but will postpone major economic damage. Stage II, which is a minimum of 17 hours, is a warning that assumes a forthcoming flood above grade. The time limits on the stages are given so that unnecessary economic penalty can be avoided while adequate time is allowed for preparing for operation in the flood mode.

Stage I procedures consist of a controlled reactor shutdown and other easily revokable steps, such as moving flood supplies above the maximum possible flood elevation and making temporary connections and load adjustments on the onsite power supply. After unit shutdown, the Reactor Coolant System will be cooled and the pressure will be reduced to less than 350 psig. Stage II procedures are the least easily revokable and more damaging steps necessary to have the plant in the flood mode when the flood exceeds plant grade. Heat removal from the steam generators will be accomplished by adding river water

(continued)

ACTIONS (continued)

with the prospect of reaching elevation of 727 ft MSL, 1 foot below plant grade, are early enough to assure adequate warning time for safe plant shutdown.

B.1

If the Stage II flood warning has been issued, the Stage II flood protection plan must be initiated and completed within 17 hours or prior to the predicted flooding of the site. The Completion Time of 17 hours corresponds to the remaining hours of the 27 hour preflood preparation time after the Stage I flood warning consisting of 10 hours has expired, and is an adequate time period to complete Stage II preparations.

C.1, C.2.1, C.2.2.1, and C.2.2.2

If a seismic event occurs after a critical combination of flood and/or headwater alerts is issued, within 6 hours communications between TVA Power Control Center and the Watts Bar Nuclear Plant must be verified and maintained.The TVA Power Control Center is able to detect unexplained electrical interruptions at dams (not including Fontana Dam), or loss of contact with the dams involved in the issued alert. If an unexplained interruption occurs, the Watts Bar Plant Manager will be notified and efforts will be made by the TVA Power Control Center to determine whether dam failure has occurred. The 6-hour Completion Time is an adequate time period to complete the requirements of Required Action C.1.

If Required Action C.1 and the associated Completion Time is not met, the Stage I flood protection plan must be initiated and completed within the 16 hours. The Completion Time for this Required Action is 16 hours which is adequate time for preparing for operation in the flood mode.

Also, communications between the TVA Power Control Center and Watts Bar Nuclear Plant must be established prior to the completion of the Stage I flood protection plan. If communications cannot be established, the Stage II flood protection plan must be initiated and completed within 17 additional hours (33 hours total). The Completion Time of 33 hours corresponds the TVA Division of Water Resources preflood preparation time and is an adequate time period to complete shutdown.

(continued)

BASES

BACKGROUND (continued)	The general guidelines, which are followed for the qualification of electrical equipment, are provided in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 1). Detailed requirements for the implementation of the general guidelines are provided in various Regulatory Guides and IEEE Standards. Basic requirements for the qualification of mechanical equipment are outlined in General Design Criteria 4 (Ref. 2).
APPLICABLE SAFETY ANALYSIS	Certain components, which have the service temperatures controlled by this requirement, are part of the primary success path and function to mitigate DBAs and Transients. However, the integrity/OPERABILITY of these components is addressed in the relevant specifications that cover individual components. The service temperatures and the thermal aging, which are controlled by observing the requirements of this TR, are not inputs to the safety analysis. Further, Probabilistic Risk Assessment studies performed to date, do not explicitly model the function of area temperature monitors. In addition, this particular requirement covers only service temperatures and thermal aging of these components, which are not considerations in designing the accident sequences for theoretical hazard evaluations (Ref. 3).
TR	TR 3.7.5 provides nominal temperature limits in the vicinity of major equipment. The TR allows for each area shown in Table 3.7.5-1 to be higher or lower than the normal limit for a maximum of eight hours.
APPLICABILITY	The limits on temperature and time apply whenever the affected equipment in an affected area is required to be OPERABLE.

(continued)

BASES (continued)

ACTIONS

A.1

Whenever the temperature in one or more areas have exceeded the normal temperature limits for more than eight hours, a report must be prepared and submitted to the NRC within 30 days. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation, and prepare the report.

Condition A has been modified by a Note stating that the provisions of TR 3.0.3 and TR 3.0.4 do not apply.

B.1.1, B.1.2, and B.2

Whenever the temperature in one or more areas exceed the abnormal temperature limits, the temperature must be restored to within the normal limits in 4 hours. The Completion Time of 4 hours is based on operator experience and is a reasonable time for restoring the temperature. Alternatively, the affected equipment must be declared inoperable. Within 30 days a report must be prepared and submitted to NRC. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation and prepare the report.

C.1 and C.2

Whenever the temperature in the Intake Pumping Station mechanical or electrical equipment rooms exceeds the lower limit of 40 °F, actions must be initiated within 24 hours to ensure the temperature does not decrease below 32 °F. The Completion Time of 24 hours is based on temperature analysis. Within 7 days, restore normal temperatures within the areas affected. The 7 day Completion Time is based on a reasonable repair duration, and compensatory actions available during the interim period to maintain temperatures above 32 °F.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the temperature in the Intake Pumping Station mechanical or electrical equipment rooms decrease to 32 °F or lower, the affected equipment must be immediately declared inoperable. The Completion Time is based on potential freezing of safety-related components. Within 30 days a report must be prepared and submitted to NRC. The report must contain the cumulative time and amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation and prepare the report.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.5.1

The temperature in each area must be determined every 12 hours to ensure compliance with the limits. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

REFERENCES

1. 10 CFR 50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants".
2. 10 CFR 50 Appendix A, General Design Criteria 4, "Environmental and Dynamic Effects Design Bases".
3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Communications

BASES

BACKGROUND	During CORE ALTERATIONS communication ability must be retained between the control room and personnel on the refueling station. This is needed to allow the refueling personnel to be informed of any significant changes in the unit status or core reactivity conditions.
APPLICABLE SAFETY ANALYSES	This requirement helps assure direct communications between the control room and refueling personnel during refueling, which would help to preclude inadvertent criticality. It also ensures that the refueling personnel are able to inform the control room if there are any problems or accidents during the refueling process. Refueling operations are not addressed in PRA studies and would not be important in accident sequences that are commonly found to dominate risk (Ref. 1).
TR	TR 3.9.2 requires that direct communications be maintained between the control room and personnel at the refueling station. This ensures that information can be exchanged between the two groups if any unplanned events occur or if any significant changes occur in the unit status or core reactivity conditions.
APPLICABILITY	TR 3.9.2 is only applicable during CORE ALTERATIONS (MODE 6). In all other MODES refueling procedures do not take place and are therefore not applicable.

(continued)