Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-15-177

September 4, 2015

10 CFR §50.36(a)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
Construction Permit No. CPPR-92
NRC Docket No. 50-391

Subject: **Watts Bar Nuclear Plant Unit 2 – Submittal of Replacement Pages for Developmental and Final Revision J of the Technical Specification & Technical Specification Bases, and Developmental and Final Revision E of the Technical Requirements Manual & Technical Requirements Manual Bases**

Reference: TVA Letter to NRC, "Watts Bar Nuclear Plant Unit 2 – Submittal of Developmental and Final Revision J of the Technical Specification & Technical Specification Bases, and Developmental and Final Revision E of the Technical Requirements Manual & Technical Requirements Manual Bases," dated July 6, 2015 [ML15187A461]

The purpose of this letter is to provide replacement pages for changes to the Technical Specification (TS) Revision J and Technical Requirements Manual (TRM) Revision E that have not been previously submitted to the Nuclear Regulatory Commission (NRC). The referenced letter submitted TS Developmental Revision J and the associated TS Bases and TRM Developmental Revision E and associated Bases. Several unresolved issues necessitated revisions to the information provided in the referenced letter.

The enclosure provides the summary of changes to Developmental Revision J and E of the TS and TRM, respectively. The enclosure provides a basis for the new replacement pages. Several of these changes have been previously submitted to the staff. These changes are also summarized in the enclosure. Attachments to the enclosure provide the new replacement pages for the TS and TS Bases and for the TRM and TRM Bases.

There are no new regulatory commitments contained in this submittal. Please contact Gordon Arent at 423-365-2004 if there are questions regarding this submittal.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 4th day of September 2015.

Respectfully,

J. W. Shea
Digitally signed by J. W. Shea
DN: cn=J. W. Shea, o=Tennessee Valley Authority, ou=Nuclear Licensing, email=jwshea@tva.gov, c=US
Date: 2015.09.04 21:16:44 -04'00'

J. W. Shea
Vice President, Nuclear Licensing

Enclosure: Summary of Technical Specification and Technical Requirements Manual Changes

cc (Enclosure):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Watts Bar Nuclear Plant, Unit 2
NRC Project Manager – Watts Bar Nuclear Plant, Unit 2

ENCLOSURE
Watts Bar Nuclear Plant Unit 2

Summary of Technical Specification and Technical Requirements Manual Changes

Tennessee Valley Authority (TVA) submitted a complete set of Technical Specifications (TS), TS Bases, Technical Requirements Manual (TRM) and TRM Bases to the Nuclear Regulatory Commission (NRC) on July 6, 2015 by Reference 1. A limited number of changes have been identified since Reference 1 was submitted that change information previously provided.

Previously Submitted Replacement Pages

By letter dated August 13, 2015, TVA provided replacement pages for TS Surveillance Requirements 3.6.11.2 and 3.6.11.3 regarding the amount of ice that must be maintained in the ice condenser. The mass of ice required was increased to support a revised containment loss of coolant accident pressure analysis that was submitted in Reference 2. These pages are not being resubmitted in this letter.

Previously Described Changes

The NRC provided recommended changes to TS Bases 3.4.17, "SG Tube Integrity." By letter dated September 3, 2014 (Reference 3), TVA agreed to make the changes and to provide TS Bases 3.4.17 pages showing the changes. These changes were not incorporated in the TS Developmental Revision J submittal. The appropriate changes are provided in Attachment 1 to this enclosure.

A proposed Environmental Protection Plan (TS Appendix B) was submitted by letter with WBN Unit 2 Developmental Revision H (Reference 4). The proposed Environmental Protection Plan (EPP) has been revised to address questions from the NRC staff provided in Reference 4 and a subsequent telephone conference between NRC and TVA staff on March 11, 2015. As discussed in the telephone conference, text from the WBN Unit 1 EPP Section 4.2, "Unusual or Important Environmental Events," has been incorporated into the WBN Unit 2 EPP Section 4.1.1, "Unusual or Important Environmental Events," in its entirety. Also as discussed during the telephone conference, Section 4.1.2 of the WBN Unit 1 EPP describing the maintenance of transmission line corridors is not included in the WBN Unit 2 EPP. The WBN Unit 2 EPP is provided in Attachment 2.

Changes Identified Since July 2015 Resulting in New Pages

The resolution of the General Design Criteria (GDC) 5 issue has produced the largest number of changes to the TS and TS Bases. New TS 3.7.16, "Component Cooling System (CCS) - Shutdown" and TS 3.7.17, "Essential Raw Cooling Water (ERCW) System - Shutdown" and the associated TS Bases were developed as part of this effort. These new TS and TS Bases sections were submitted on WBN Unit 1 as a license amendment request (Reference 6). The NRC conducted an audit of the WBN Unit 1 license amendment in support of resolution of GDC 5 issues. The questions and issues that the staff raised during the audit were addressed in Reference 7. Resolution of the GDC 5 issues resulted in modifications to the submitted CCS and ERCW TS and associated TS Bases.

TVA proposed to make additional changes to existing TS in response to discussions during the GDC 5 audit. A change has been made to the Mode of Applicability for Limiting Condition for Operation 3.3.2, "Engineered Safety Feature Actuation Instrumentation," Function 6.f related to switching the source of water for the Auxiliary Feedwater System from the Condensate Storage

Tank to the Essential Raw Cooling Water (ERCW) System. The table has been updated to note that this function is required to be operable in Mode 4 when the steam generators are being used for heat removal. The TS Bases for 3.3.2 Function 6.f was revised to describe the change in Mode applicability. Changes have been made to TS 3.4.6, "RCS Loops - MODE 4," to add a requirement that the unit may not proceed to Mode 5 until at least seven hours have elapsed after entry into Mode 3 from Mode 1 or 2. These changes also require a minimum of two Reactor Coolant System (RCS) loops be operable, with one loop in operation for the first seven hours after entry into Mode 3 from Mode 1 or 2. This requirement is irrespective of the number of Residual Heat Removal Loops that are available or in operation.

Attachment 3 provides new TS and TS Bases pages for TS 3.3.2, 3.4.6, 3.7.16 and 3.7.17 to reflect the resolution of the GDC 5 audit issues.

Technical Requirements Manual Figure 3.1.6, "Boric Acid Tank Limits," has been revised. The figure submitted in TRM Developmental Revision E was the same figure that is in the WBN Unit 1 TRM. The WBN Unit 2 figure was revised to account for Unit 2 having the original steam generators, resulting in a smaller primary system volume than WBN Unit 1 has with the replacement steam generators. The revised figure is provided in Attachment 4 to this enclosure.

References:

1. TVA Letter to NRC, "Watts Bar Nuclear Plant Unit 2 – Submittal of Developmental and Final Revision J of the Technical Specification & Technical Specification Bases, and Developmental and Final Revision E of the Technical Requirements Manual & Technical Requirements Manual Bases," dated July 6, 2015 [ML15187A461]
2. TVA Letter to NRC, "Revised FSAR Section 6.2.1 Containment Functional Design," dated August 13, 2015 [ML115225A382]
3. TVA Letter to NRC, "Watts Bar Nuclear Plant Unit 2 – Response to NRC Requests Related to Granting an Operating License," dated September 3, 2014 [ML14246A546]
4. TVA Letter to NRC, "Watts Bar Nuclear Plant Unit 2 – Submittal of Developmental Revision H of the Unit 2 Technical Specification and Technical Specification Bases," dated December 12, 2013 [ML13357A048]
5. NRC Letter to TVA, "Watts Bar Nuclear Plant, Unit 2 - Environmental Protection Plan (Non-Radiological) Technical Specification Review," dated February 2, 2015 [ML15015A477]
6. TVA Letter to NRC, "Watts Bar Nuclear Plant Unit 1 - Application to Revise Technical Specifications for Component Cooling Water and Essential Raw Cooling Water to Support Dual Unit Operation (TS-WBN-15-13)," dated June 17, 2015 [ML15170A474]
7. TVA Letter to NRC, "Responses to NRC Audit Review Questions for Watts Bar Nuclear Plant Unit 1 Essential Raw Cooling Water and Component Cooling Water System License Amendment Request," dated August 28, 2015 [ML15243A044]

Attachment 1
Watts Bar Nuclear Plant Unit 2

Technical Specification Bases
3.4.17, SG Tube Integrity - Marked-up

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed 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," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.7.2.12, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.7.2.12, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.7.2.12. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, “RCS Operational LEAKAGE,” plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than an SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE ~~from~~ of 150 gallons per day (gpd) per unfaulted steam generator and 1 gallon per minute (gpm) in the faulted steam generator. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, “RCS Specific Activity,” limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), and 10 CFR 100 (Ref. 3) or the NRC approved licensing basis.

Steam generator tube integrity satisfies Criterion 2 of the NRC Commission Interim Policy Statement (Ref. 7).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

(continued)

LCO (continued)

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.7.2.12, “Steam Generator Program,” and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, “The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.” Tube collapse is defined as, “For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero.” The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term “significant” is defined as “An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established.” For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions), and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

(continued)

BASES

LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than an SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm in the faulted SG. The accident induced leakage rate includes any primary-to-secondary LEAKAGE existing prior to the accident in addition to primary-to-secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary-to-secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to an SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry, and application of associated Required Actions.

(continued)

ACTIONS (continued)

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if an SG tube that should have been plugged , has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.7.2.12 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.7.2.12 until subsequent inspections support extending the inspection interval.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 5.7.2.12 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

~~Steam Generator tube plugging is only performed using approved plugging methods as described in the Steam Generator Program.~~

The Frequency of prior to entering MODE 4 following an SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

REFERENCES

1. NEI 97-06, “Steam Generator Program Guidelines.”
2. 10 CFR 50 Appendix A, GDC 19, Control Room.
3. 10 CFR 100, Reactor Site Criteria.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.
6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines.”

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REFERENCES (continued)

7. NRC Commission, “Interim Staff Policy Statement on Technical Specification Improvements for Nuclear Power Reactors,” Federal Register 52 FR 3788, dated February 6, 1987; Westinghouse Owners Group (R.A. Newton) letter to NRC Document Control Desk, “Westinghouse Owners Group MERITS Program Phase II, Task 5, Criteria Application Topical Report,” dated November 12, 1987; NRC (T.E. Murley to W.S. Wilgus) letter, “NRC Staff Review of Nuclear Steam Supply System Vendor Owners, Groups’ Application of the Commission’s Interim Policy Statement Criteria to Standard Technical Specification,” dated May 9, 1988, ADAMS Accession No. ML11264A057; TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Proposed Technical Specifications (TS),” dated August 27, 1992, ADAMS Accession No. ML073200281; and NRC letter, “Issuance of Facility Operating License No. NPF-90, Watts Bar Nuclear Plant, Unit 1(TAC M94025),” dated February 7, 1996, ADAMS Accession No. ML052930169.

Attachment 1
Watts Bar Nuclear Plant Unit 2

Technical Specification Bases
3.4.17, SG Tube Integrity - Clean

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed 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," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.7.2.12, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.7.2.12, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.7.2.12. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, “RCS Operational LEAKAGE,” plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than an SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE of 150 gallons per day (gpd) per unfaulted steam generator and 1 gallon per minute (gpm) in the faulted steam generator. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, “RCS Specific Activity,” limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), and 10 CFR 100 (Ref. 3) or the NRC approved licensing basis.

Steam generator tube integrity satisfies Criterion 2 of the NRC Commission Interim Policy Statement (Ref. 7).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

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LCO (continued)

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.7.2.12, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions), and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

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BASES

LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than an SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm in the faulted SG. The accident induced leakage rate includes any primary-to-secondary LEAKAGE existing prior to the accident in addition to primary-to-secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, “RCS Operational LEAKAGE,” and limits primary-to-secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to an SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry, and application of associated Required Actions.

(continued)

ACTIONS (continued)

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if an SG tube that should have been plugged , has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.7.2.12 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.7.2.12 until subsequent inspections support extending the inspection interval.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 5.7.2.12 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following an SG inspection ensures that the Surveillance has been completed and all tubes meeting the plugging criteria are plugged prior to subjecting the SG tubes to significant primary-to-secondary pressure differential.

REFERENCES

1. NEI 97-06, “Steam Generator Program Guidelines.”

2. 10 CFR 50 Appendix A, GDC 19, Control Room.

3. 10 CFR 100, Reactor Site Criteria.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.

6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines.”

(continued)

BASES

REFERENCES (continued)

7. NRC Commission, “Interim Staff Policy Statement on Technical Specification Improvements for Nuclear Power Reactors,” Federal Register 52 FR 3788, dated February 6, 1987; Westinghouse Owners Group (R.A. Newton) letter to NRC Document Control Desk, “Westinghouse Owners Group MERITS Program Phase II, Task 5, Criteria Application Topical Report,” dated November 12, 1987; NRC (T.E. Murley to W.S. Wilgus) letter, “NRC Staff Review of Nuclear Steam Supply System Vendor Owners, Groups’ Application of the Commission’s Interim Policy Statement Criteria to Standard Technical Specification,” dated May 9, 1988, ADAMS Accession No. ML11264A057; TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Proposed Technical Specifications (TS),” dated August 27, 1992, ADAMS Accession No. ML073200281; and NRC letter, “Issuance of Facility Operating License No. NPF-90, Watts Bar Nuclear Plant, Unit 1(TAC M94025),” dated February 7, 1996, ADAMS Accession No. ML052930169.

(continued)

Attachment 2
Watts Bar Nuclear Plant Unit 2

Technical Specifications Appendix B
Environmental Protection Plan

APPENDIX B

TO FACILITY OPERATING LICENSE

ENVIRONMENTAL PROTECTION PLAN
(NON-RADIOLOGICAL)

FOR

WATTS BAR NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-391

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT

UNIT 2

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

TABLE OF CONTENTS

1.0 DEFINITIONS, ABBREVIATIONS, AND ACRONYMS

Annually	Annually is once per calendar year at intervals of twelve (12) calendar months ± 30 days
Clean Water Act	Federal Water Pollution Control Act (FWPCA) as amended.
FES	Final Environmental Statement (NUREG-0498) issued December 1978 by the NRC to the TVA (Control No. 7901100061).
FES Supplement 1	Final Environmental Statement (NUREG-0498 Supplement 1) issued April 1995 by the NRC to the TVA (ADAMS Accession No. ML081430592).
FES Supplement 2	Final Environmental Statement (NUREG-0498 Supplement 2, Vol. 1 & Vol. 2) issued May 2013 by the NRC to the TVA (ADAMS Accession Nos. ML13144A092 & ML13144A093).
FWS	U.S. Fish and Wildlife Service
NPDES Permit	NPDES permit is the National Pollutant Discharge Elimination System Permit No. TN0020168 issued by the U.S. Environmental Protection Agency to the Tennessee Valley Authority (TVA). This permit authorizes TVA to discharge controlled waste water, from the Watts Bar Plant Unit 2 into the Tennessee River.
NRC	U.S. Nuclear Regulatory Commission
Plant	Plant refers to the Watts Bar Nuclear Plant, either Unit 1 or Unit 2.
Site	Onsite includes any area within the property owned by the TVA specifically described in the WBN FES. Offsite includes all other areas.
Station	Station refers to Watts Bar Nuclear Plant Unit 1 and Unit 2.
TVA	Tennessee Valley Authority
Unit	Unit refers to Unit 1 or 2 (i.e., WBN Unit 1 or WBN Unit 2) of the Watts Bar Nuclear Plant, as defined by its usage.
WBN	Watts Bar Nuclear Plant

2.0 LIMITING CONDITIONS FOR OPERATION (N/A)

None required

3.0 ENVIRONMENTAL MONITORING[1]

Environmental monitoring programs are conducted in accordance with the guidance and controls of agencies outside of the NRC. The NRC will relay on decisions made by the U.S. Environmental Protection Agency, U.S. Fish and Wildlife Service, and the State of Tennessee for any requirements on environmental monitoring. Therefore, no specific environmental monitoring is required by the NRC under this EPP.

3.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decision made by the U.S. Environmental Protection Agency and the State of Tennessee under the authority of the Clean Water Act for any requirements for aquatic monitoring.

3.2 Terrestrial Monitoring

Terrestrial monitoring is not required.

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[1] In consideration of the provisions of the Clean Water Act (33 USC §1251, et seq.) and in the interest of avoiding duplication of effort, the conditions and monitoring requirements related to water quality and aquatic biota are specified in the National Pollution Discharge Elimination System (NPDES) Permit No. TN0020168 issued by the U.S. Environmental Protection Agency to the Tennessee Valley Authority (TVA). This permit authorizes TVA to discharge controlled waste water from the Watts Bar Nuclear Plant Unit 2 into the Tennessee River.

The Nuclear Regulatory Commission will be relying on the NPDES permit for protection of the aquatic environment from non-radiological effluents.

4.0 SPECIAL STUDIES AND REQUIREMENTS

4.1 Exceptional Occurrences

4.1.1 Unusual or Important Environmental Events

Requirements

Any occurrence of an unusual event or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report in accordance with Subsection 5.4.2. If an event is reportable under 10 CFR 50.72, then a duplicate immediate report under this subsection is not required. However, a follow-up written report is required in accordance with Subsection 5.4.2.

The following are examples: excessive bird impact events, onsite plant or animal disease outbreaks, mortality of, or unusual occurrence involving any species protected by the Endangered Species Act of 1973 (ESA), the identification of any threatened or endangered species for which the NRC has not initiated consultation with the FWS, fish kills, increase in nuisance organisms or conditions in excess of levels anticipated in station environmental impact appraisals, and unanticipated or emergency discharge of waste water or any other chemical substance that exceeds the limits of, or is not authorized by, the NPDES permit and requires 24-hour notification to the State of Tennessee.

The licensee shall also notify the FWS Cookeville Field Office Field Supervisor or his designee when an unusual or important event results in the taking of, or could result in an adverse impact to, any species protected by the ESA. TVA should also notify the FWS law enforcement agent in Nashville, Tennessee if an unusual or important event involves the death, injury, or illness of any individual of a species protected by the ESA. Initial notification must be completed within 24 hours of the unusual or important event, followed by a written report per subsection 5.4.2

No routine monitoring programs are required to implement this condition.

Action

Should an “Unusual or Important Environmental Event” occur, the licensee shall make a prompt report to the NRC in accordance with the provisions of Subsections 5.4.2.a and 5.4.2.c, or Subsection 5.4.2.d.

4.1.2 Exceeding Limits of Other Relevant Permits

Requirement

The licensee shall notify the NRC of occurrences in which the limits specified, in relevant permits and certificates issued by other Federal, State, and local governments are exceeded and which are reportable to those agencies. This requirement shall commence with the date of issuance of the operating license and continue for the life of the plant, unless changed in accordance with Subsection 5.5.1.

Action

The licensee shall make a report to the NRC in accordance with the provisions of Subsections 5.4.2.b and 5.4.2.c, or Subsection 5.4.2.d in the event of a reportable occurrence in which a limit specified in a relevant permit or certificate issued by another Federal, State, or local agency is exceeded.

4.2 Special Studies

None required at the present time.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

The Plant Manager has responsibility for operating the plant in compliance with this Environmental Protection Plan.

5.2 Review and Audit

The licensee shall provide for review and audit of compliance with the Environmental Protection Plan. The audits shall be conducted independently of the Individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2.1 Review

The licensee is responsible for the review of procedures for meeting the Environmental Protection Plan.

The above mentioned review shall be conducted on the following:

A. Proposed changes to the Environmental Protection Plan and evaluated impact of the change.

B. Proposed changes to station operating procedures, which affect the environmental effects of the station.

C. Proposed changes, construction, or modifications to station or unit equipment, or systems which might have an environmental impact, in order to determine the environmental impact of the change[2].

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[2] Activities are excluded from this requirement if all measurable environmental effects are confined to on-site areas previously disturbed during site preparation and plant construction.

D. All routine reports prior to their submittals to NRC (described in Subsection 5.4.1).

E. All nonroutine reports prior to submittal of the written report (described in Subsections 5.4.a, b, and c).

F. Investigations of all reported instances of noncompliance with the Environmental Protection Plan, associated corrective actions, and measures taken to prevent recurrence.

5.2.2 Audit

The licensee shall conduct an audit on the environmental monitoring program. The audits shall be conducted independently of the individual or group responsible for performing the specific activity. Results of the audit activities shall be maintained and made available for inspection.

5.3 Changes in Station Design or Operation

Changes in station design or operation may be made subject to the following conditions:

A. The licensee may (1) make changes in the station design and operation, and (2) conduct tests and experiments not described in this document without prior Commission approval, unless the proposed change, test or experiment involves a change in the objectives of the Environmental Protection Plan[3] and/or an unreviewed environmental question of significant impact.

B. A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the final environmental impact statement as modified by testimony to the Atomic Safety and Licensing Board, supplements thereto, environmental impact appraisals, or in initial or final adjudicatory decisions; or (2) a matter not previously reviewed and evaluated in the documents specified in (1) of this section which may have a significant adverse environmental impact.

C. The licensee shall maintain records of changes in facility design or operation made pursuant to this subsection. The licensee shall also maintain records of tests ad experiments carried out pursuant to paragraph “A” of this Subsection. These records shall include a written change, test, or experiment does not involve an unreviewed environmental question or substantive impact or constitute a change in the objectives of the Environmental Protection Plan. The licensee shall furnish to the Commission, annually or at such shorter intervals as may be specified in the license, a report containing description, analyses, interpretations, and evaluations of such changes, tests, and experiments.

[3] This provision does not relieve the licensee of the requirements of 10 CFR 50.59.

D. Changes in the special studies, if required in Section 4.2, which affects sampling frequency, location, gear, or replication shall be reported to the NRC within 30 days after their implementation, unless otherwise reported in accordance with Subsection 5.4.2. These reports shall describe the changes made, the reasons for making the changes, and an evaluation of the effectiveness of the revised program in assessing environmental impacts.

5.4 Station Reporting Requirements

5.4.1 Routine Reports

Annual Environmental Operation Report

A WBN dual-unit report on the environmental monitoring program for the previous year shall be submitted to the NRC separate from other NRC reporting requirements within 90 days following each anniversary of issuance of the WBN Unit 1 operating license. The WBN Unit 1 operating license anniversary date is utilized as the basis for the WBN dual-unit anniversary date, since it was the basis for the initial and subsequent reports. The report shall include summaries, analyses, interpretations, and statistical evaluation of the results of the environmental monitoring required by special studies and requirements (Section 4) for the report period, including a comparison with preoperational studies, operating controls (as appropriate), and previous non-radiological environmental monitoring reports, and an assessment of the observed impacts of the station operation on the environment. If harmful effects or evidence of irreversible damage are suggested by the monitoring programs, the licensee shall provide a more detailed analysis of the data and a proposed course of action to alleviate the problem.

For those programs concerned with water quality or protection of aquatic biota, which are regulated under the Clean Water Act, the requirements of this section shall be satisfied by submitting to the NRC copies of the reports as required by the NPDES permit (or otherwise required pursuant to the Clean Water Act), and in accordance with the frequency, content and schedules set forth by the agencies responsible for implementing the Clean Water Act.

In the event that some results are not available by the report date, the report shall be submitted noting and explaining the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The Annual Report shall also include a summary of:

1. All Environmental Protection Plan noncompliances and the corrective actions taken to remedy them.
2. Changes made to applicable State and Federal permits and certifications.
3. Changes to station design which could involve a significant environmental impact or change the findings of the FES.
4. All nonroutine reports submitted per Environmental Protection Plan Section 4.1.
5. Changes in the approved Environmental Protection Plan.

5.4.2 Nonroutine Reports

A report shall be submitted in the event that an "Unusual or Important Environmental Event," as specified in Subsection 4.1.1 occurs, or if another relevant permit is violated as specified in Subsection 4.1.2. The schedule and content for these nonroutine reports are described below.

5.4.2.a Prompt Report

Those events specified as requiring prompt reporting shall be reported within 24 hours by telephone, telegraph, or facsimile transmission to the NRC followed by a written report to the NRC within 30 days.

5.4.2.b Thirty Day Report

Those events not requiring a prompt report as described in Subsection 5.4.2.a shall be reported to the NRC within 30 days of their occurrence.

5.4.2.c Content of Nonroutine Reports

Written 30-day reports and, to the extent possible, the preliminary telephone, telegraph, or facsimile reports shall (a) describe, analyze, and evaluate the occurrence, including extent and magnitude of the impact, (b) describe the cause of the occurrence, (c) indicate the action taken to correct the reporting occurrence, and (d) indicate the corrective action taken (including any significant changes made in procedures) to preclude repetition of the occurrence and to prevent similar occurrences involving similar components or systems.

5.4.2.d Exceptions for Matters Regulated Under the Clean Water Act

For matters regulated under the Clean Water Act, the report schedules and content requirements described in Subsections 5.4.2.a, 5.4.2.b, and 5.4.2.c shall be satisfied by submitting, to the NRC copies of the reports as required b the NPDES permit (or other regulations pursuant to the Clean Water Act) and in accordance with the schedules and content requirements imposed thereby.

5.5 Changes in the Environmental Protection Plan and Permits

5.5.1 Changes in the Environmental Protection Plan

Requests for change to the Environmental Protection Plan shall be submitted to the NRC for review and authorization per 10 CFR 50.90. The request shall include an evaluation of the environmental impact of the proposed change and a supporting justification. Implementation of such requested changes to the Environmental Protection Plan shall not commence prior to incorporation by the NRC of the specifications in the license.

5.5.2 Changes in Permits and Certifications

Changes and additions to required Federal (other than NRC), State, local, and regional authority permits and certificates for the protection of the environment shall be reported to the NRC within 30 days. In the event that the licensee initiates or becomes aware of a request for changes to any of the water quality requirements, limits, or values stipulated in any certification or permit issued pursuant to the Clean Water Act, the NRC shall be notified within 30 days.

If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days. If, as a result of the appeal process, the permit or certification requirements are changed, the change shall be dealt with as described in the previous paragraph of this section.

5.6 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

5.6.1 The following records shall be retained for the life of the station:

(a) Record of changes to the Environmental Protection Plan including, when applicable, records of NRC approval of such changes.

(b) Record of modifications to station structures, systems, and components determined to potentially affect the continued protection of the environment.

(c) Record of changes to permits and certifications required by Federal (other than the NRC), State, local, and regional authorities for the protection of the environment.

(d) Routine reports submitted to the NRC.

5.6.2 Records of the following shall be retained for a minimum of six (6) years:

(a) Review and audit activities.

(b) Events, and the reports thereon, which are the subjects of non-routine reports to the NRC.

5.6.3 Records associated with requirements of Federal (other than the NRC), State, local, and regional authorities' permits and certificates for the protection of the environment shall be retained for the period established by the respective permit or certificate.

Attachment 3

Watts Bar Nuclear Plant Unit 2

Technical Specifications and Technical Specification Bases

Technical Specification and Technical Specification Bases 3.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation”

- Function 6.f, “Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure - Low” - Marked-up

Technical Specification and Technical Specification Bases 3.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation”

- Function 6.f, “Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure - Low” - Clean

Technical Specification and Technical Specification Bases 3.4.6, “RCS Loops - MODE 4” - Marked-up

Technical Specification and Technical Specification Bases 3.4.6, “RCS Loops - MODE 4” - Clean

Technical Specification and Technical Specification Bases 3.7.16, “Component Cooling System (CCS) - Shutdown” - New

Technical Specification and Technical Specification Bases 3.7.17, “Essential Raw Cooling Water (ERCW) System - Shutdown” - New

Table 3.3.2-1 (page 6 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
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. Notes (b) and (c) are applicable to SR 3.3.5.2 for this function.		
e. Trip of all Turbine Driven Main Feedwater Pumps	1 (j), 2 (k)	1 per pump	J	SR 3.3.2.8 (b)(c) SR 3.3.2.9 (b) (c) SR 3.3.2.10	≥43.3 psig	50 psig
f. Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure - Low	1, 2, 3, 4(m)	3	B~~F~~	SR 3.3.2.6 SR 3.3.2.9 (b) (c) SR 3.3.2.10	A) ≥ 0.5 psig B) ≥ 1.33 psig	A) 1.2 psig B) 2.0 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
b. Refueling Water Storage Tank (RWST) Level - Low	1, 2, 3, 4	4	K	SR 3.3.2.1 SR 3.3.2.4 (b) (c) SR 3.3.2.9 (b) (c) 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 (b) (c) SR 3.3.2.9 (b) (c) SR 3.3.2.10	≥ 37.2 inches above el. 702.8 ft	38.2 inches above el. 702.8 ft

(continued)

(b) If the as found channel setpoint is outside its redefined as found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The methodologies used to determine the as found and as left tolerances for the NTSP are specified in FSAR Section 7.1.2.

(j) Entry into Condition J may be suspended for up to 4 hours when placing the second Turbine Driven Main Feedwater (TDMFW) Pump in service or removing one of two TDMFW pumps from service.

(k) When one or more Turbine Driven Feedwater Pump(s) are supplying feedwater to steam generators.

(m) When steam generators are relied on for heat removal.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

f. <u>Auxiliary Feedwater - Pump Suction Transfer on Suction Pressure - Low</u>

A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. Three pressure switches are located on each motor driven AFW pump suction line from the CST. A low pressure signal sensed by two switches of a set will cause the emergency supply of water for the respective pumps to be aligned. ERCW (safety grade) is then lined up to supply the AFW pumps to ensure an adequate supply of water for the AFW System to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.

Since the detectors are located in an area not affected by HELBs or high radiation, they will not experience any adverse environmental conditions and the NTSP reflects only steady state instrument uncertainties.

This~~ese~~ Function~~s~~ must be OPERABLE in MODES 1, 2, ~~and~~ 3, and 4, when the steam generators are relied on to remove decay heat from the reactor, to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. This~~ese~~ Function~~s~~ does not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

(continued)

BASES

ACTIONS (continued)

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

<u>A.1</u>

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

<u>B.1, B.2.1 and B.2.2</u>

Condition B applies to manual initiation of:

- SI;
- Containment Spray;
- Phase A Isolation; and
- Phase B Isolation.

Condition B also applies to the Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low.

For the manual initiation Functions, t~~T~~his action addresses the train orientation of the SSPS for the functions listed above. For the AFW System pump suction transfer channels, this action recognizes that placing a failed channel in trip during operation is not necessarily a conservative action. Spurious trip of this function could align the AFW System to a source that is not immediately capable of supporting pump suction. If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. Note that for containment spray and Phase B isolation, failure of one or both channels in one train renders the train inoperable. Condition B, therefore, encompasses both situations.

(continued)

BASES

ACTIONS

B.1, B.2.1 and B.2.2 (continued)

For the manual initiation Functions, ~~T~~the specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each Function, and the low probability of an event occurring during this interval. For the AFW System pump suction transfer channels, the specified Completion Time is reasonable considering the nature of this Function, the available redundancy, and the low probability of an event occurring during this interval. If the channel or train cannot be restored to OPERABLE status, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 30 hours (84 hours total time). The allowable 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. For the AFW System pump suction transfer channels, aligning the RHR System for decay heat removal, so that the steam generators are not relied on for heat removal, places the plant in a MODE in which the LCO no longer applies. Therefore, per LCO 3.0.2, completion of the Required Action to place the unit in MODE 5 is not required.

For the manual initiation Functions, ~~T~~the allowance of 48 hours is justified in Reference 7.

(continued)

BASES

ACTIONS (continued)

F.1, F.2.1, and F.2.2

Condition F applies to:

- Manual Initiation of Steam Line Isolation;
- ~~•~~ Loss of Offsite Power;
- ~~Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low;~~ and
- P-4 Interlock.

For the Manual Initiation and the P-4 Interlock Functions, this action addresses the train orientation of the SSPS. For the Loss of Offsite Power Function, this action recognizes the lack of manual trip provision for a failed channel. ~~For the AFW System pump suction transfer channels, this action recognizes that placing a failed channel in trip during operation is not necessarily a conservative action. Spurious trip of this function could align the AFW System to a source that is not immediately capable of supporting pump suction.~~ If a train or channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of these Functions, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the plant must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and without challenging plant systems. In MODE 4, the plant does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

(continued)

Table 3.3.2-1 (page 6 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
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. Notes (b) and (c) are applicable to SR 3.3.5.2 for this function.		
e. Trip of all Turbine Driven Main Feedwater Pumps	1 [j], 2 [k]	1 per pump	J	SR 3.3.2.8 [b][c] SR 3.3.2.9 [b][c] SR 3.3.2.10	≥43.3 psig	50 psig
f. Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure - Low	1, 2, 3, 4[m]	3	B	SR 3.3.2.6 SR 3.3.2.9 [b][c] SR 3.3.2.10	A) ≥ 0.5 psig B) ≥ 1.33 psig	A) 1.2 psig B) 2.0 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
b. Refueling Water Storage Tank (RWST) Level - Low	1, 2, 3, 4	4	K	SR 3.3.2.1 SR 3.3.2.4 [b][c] SR 3.3.2.9 [b][c] 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 [b][c] SR 3.3.2.9 [b][c] SR 3.3.2.10	≥ 37.2 inches above el. 702.8 ft	38.2 inches above el. 702.8 ft

(continued)

(b) If the as found channel setpoint is outside its redefined as found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The methodologies used to determine the as found and as left tolerances for the NTSP are specified in FSAR Section 7.1.2.

(j) Entry into Condition J may be suspended for up to 4 hours when placing the second Turbine Driven Main Feedwater (TDMFW) Pump in service or removing one of two TDMFW pumps from service.

(k) When one or more Turbine Driven Feedwater Pump(s) are supplying feedwater to steam generators.

(m) When steam generators are relied on for heat removal.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

f. <u>Auxiliary Feedwater - Pump Suction Transfer on Suction Pressure - Low</u>

A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. Three pressure switches are located on each motor driven AFW pump suction line from the CST. A low pressure signal sensed by two switches of a set will cause the emergency supply of water for the respective pumps to be aligned. ERCW (safety grade) is then lined up to supply the AFW pumps to ensure an adequate supply of water for the AFW System to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.

Since the detectors are located in an area not affected by HELBs or high radiation, they will not experience any adverse environmental conditions and the NTSP reflects only steady state instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, 3, and 4, when the steam generators are relied on to remove decay heat from the reactor, to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

(continued)

ACTIONS (continued)

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1, B.2.1 and B.2.2

Condition B applies to manual initiation of:

- SI;
- Containment Spray;
- Phase A Isolation; and
- Phase B Isolation.

Condition B also applies to the Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low.

For the manual initiation Functions, this action addresses the train orientation of the SSPS for the functions listed above. For the AFW System pump suction transfer channels, this action recognizes that placing a failed channel in trip during operation is not necessarily a conservative action. Spurious trip of this function could align the AFW System to a source that is not immediately capable of supporting pump suction. If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. Note that for containment spray and Phase B isolation, failure of one or both channels in one train renders the train inoperable. Condition B, therefore, encompasses both situations.

(continued)

BASES

ACTIONS

B.1, B.2.1 and B.2.2 (continued)

For the manual initiation Functions, the specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each Function, and the low probability of an event occurring during this interval. For the AFW System pump suction transfer channels, the specified Completion Time is reasonable considering the nature of this Function, the available redundancy, and the low probability of an event occurring during this interval. If the channel or train cannot be restored to OPERABLE status, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 30 hours (84 hours total time). The allowable 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. For the AFW System pump suction transfer channels, aligning the RHR System for decay heat removal, so that the steam generators are not relied on for heat removal, places the plant in a MODE in which the LCO no longer applies. Therefore, per LCO 3.0.2, completion of the Required Action to place the unit in MODE 5 is not required.

For the manual initiation Functions, the allowance of 48 hours is justified in Reference 7.

(continued)

BASES

ACTIONS (continued)

F.1, F.2.1, and F.2.2

Condition F applies to:

- Manual Initiation of Steam Line Isolation;
- Loss of Offsite Power; and
- P-4 Interlock.

For the Manual Initiation and the P-4 Interlock Functions, this action addresses the train orientation of the SSPS. For the Loss of Offsite Power Function, this action recognizes the lack of manual trip provision for a failed channel. If a train or channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of these Functions, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the plant must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power in an orderly manner and without challenging plant systems. In MODE 4, the plant does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops shall be OPERABLE, and consist of either:

a. Any combination of RCS loops and residual heat removal (RHR) loops, and one loop shall be in operation, when the rod control system is not capable of rod withdrawal; or

b. Two RCS loops, and both loops shall be in operation, when the rod control system is capable of rod withdrawal.

--NOTES--

1. No RCP shall be started with any RCS cold leg temperature ≤ 350ºF unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures.

2. For the initial 7 hours after entry into MODE 3 from MODE 1 or MODE 2, two loops shall consist of:

    a. Two RCS loops with one loop in operation when the rod control system is not capable of rod withdrawal; or

    b. Two RCS loops with both loops in operation when the rod control system is capable of rod withdrawal.

3. Average reactor coolant temperature shall be maintained > 200ºF for the initial 7 hours after entry into MODE 3 from MODE 1 or MODE 2.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Only one RCS loop OPERABLE. <u>AND</u> Two RHR loops inoperable. <u>OR</u> Less than 7 hours since entry into MODE 3 from MODE 1 or MODE 2.	A.1	Initiate action to restore a second required RCS or RHR loop to OPERABLE status.	Immediately
B. One required RHR loop inoperable. <u>AND</u> No RCS loops OPERABLE.	B.1	Be in MODE 5.	24 hours
C. One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal.	C.1	Restore required RCS loop to operation.	1 hour
	<u>OR</u>		
	C.2	De-energize all control rod drive mechanisms (CRDMs).	1 hour
D. Required RCS or RHR loops inoperable. <u>OR</u> No required RCS or RHR loop in operation	D.1	De-energize all CRDMs.	Immediately
	<u>AND</u>		
	D.2	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u>		
	D.3	Initiate action to restore one required loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify two RCS loops are in operation when the rod control system is capable of rod withdrawal.	12 hours
SR 3.4.6.2	Verify one required RHR or RCS loop is in operation when the rod control system is not capable of rod withdrawal.	12 hours
SR 3.4.6.3	Verify SG secondary side water levels are greater than or equal to 32% narrow range for required RCS loops.	12 hours
SR 3.4.6.4	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

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

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

In MODE 4, with the reactor trip breakers open and the rods not capable of withdrawal, either RCPs or RHR loops can be used to provide forced circulation. The intent in this case is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent is to require that two paths be available to provide redundancy for decay heat removal.

In MODE 4, with the reactor trip breakers closed and the rods capable of withdrawal, two RCPs must be OPERABLE and in operation to provide forced circulation.

During a normal shutdown, decay heat removal is via the RCS loops until sometime after the unit has been cooled down to RHR entry conditions (T_{cold} < 350ºF). Therefore, as LCO 3.4.6 becomes Applicable (entry into MODE 4) the RCS loops are still OPERABLE. Transitioning decay heat removal to the RHR System will place high heat loads on the RHR System, Component Cooling System (CCS), and the Essential Raw Cooling Water System (ERCW). Residual and decay heat from the RCS is transferred to CCS via the RHR HX. Heat from the CCS is transferred to the ERCW System via the CCS HXs. The CCS and ERCW systems are common between the two operating units.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

In MODE 4, with the reactor trip breakers open and the rods not capable of withdrawal, RCS circulation is considered in determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 4 with RTBs in the closed position and Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, any combination of two RCS or RHR loops are required to be OPERABLE, but only one loop is required to be in operation to meet decay heat removal requirements, except during the initial seven hours after unit shutdown, when the decay and latent heat load may exceed the heat removal capability of one RHR loop in operation.

During the initial seven hours after reactor shutdown, the heat loads are at sufficiently high levels that the requirement of LCO 3.4.6 for one RHR loop in operation may not be sufficient to mitigate a design basis accident on Unit 1 and preclude a heatup of Unit 2.

To assure that there would be adequate heat removal capability under all postulated conditions during the initial seven hours after unit shutdown, reliance on heat removal via RCS loops is required. After a unit has been shutdown for greater than seven hours, a single RHR loop in operation provides adequate heat removal capability.

RCS Loops - MODE 4 have been identified in the NRC Policy Statement as important contributors to risk reduction.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE. In MODE 4 with the RTBs in the closed position and Rod Control System capable of rod withdrawal, two RCS loops must be OPERABLE and in operation. Two RCS loops are required to be in operation in MODE 4 with RTBs closed and Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

With the RTBs in the open position, or the CRDMs de-energized, the Rod Control System is not capable of rod withdrawal; therefore, only one loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. In this case, the LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

~~The~~ Note 1 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature less than or equal to 350ºF. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 2 requires two RCS loops to be OPERABLE during the initial seven hours after entry into MODE 3 from MODE 1 or MODE 2 until decay heat and latent heat are within the capacity of the RHR System.

Note 3 precludes entry into MODE 5 during the initial seven hours after entry into MODE 3 from MODE 2 or MODE 1. This ensures that heat removal capability via RCS loops in retained until decay heat and latent heat are within the capacity of the RHR System.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG, which has the minimum water level specified in SR 3.4.6.3.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

(continued)

BASES (continued)

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

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 3.4.7, "RCS Loops - MODE 5, Loops Filled";

LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";

LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If only one RCS loop is OPERABLE and both RHR loops are inoperable, redundancy for heat removal is lost~~.~~, and ~~A~~action must be initiated to restore a second RCS or RHR loop to OPERABLE status. If only one RCS loop is OPERABLE and it has been less than seven hours since the unit has entered MODE 3 from MODE 1 or MODE 2, redundancy has been lost and action must be initiated to restore a second RCS loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1

If one required RHR loop is OPERABLE and in operation and there are no RCS loops OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the plant must be brought to MODE 5 within 24 hours. Bringing the plant to MODE 5 is a conservative action with regard to decay heat removal. With only one required RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 (less than or equal to 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of

(continued)

BASES

ACTIONS

B.1 (continued)

24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

C.1 and C.2

If one required RCS loop is not in operation, and the RTBs are closed and Rod Control System capable of rod withdrawal, the Required Action is either to restore the required RCS loop to operation or to de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets. When the RTBs are in the closed position and Rod Control System capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the RTBs must be opened. The Completion Times of 1 hour to restore the required RCS loop to operation or de-energize all CRDMs is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2 and D.3

If no loop is OPERABLE or in operation, all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one required RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. Opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.2

This SR requires verification every 12 hours that one required RCS or RHR loop is in operation when the rod control system is not capable of rod withdrawal. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.3

SR 3.4.6.3 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is greater than or equal to 32% (value does not account for instrument error, Ref. 1). If the SG secondary side narrow range water level is less than 32%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.4

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. NRC Commission, “Interim Staff Policy Statement on Technical Specification Improvements for Nuclear Power Reactors,” Federal Register 52 FR 3788, dated February 6, 1987; Westinghouse Owners Group (R.A. Newton) letter to NRC Document Control Desk, “Westinghouse Owners Group MERITS Program Phase II, Task 5, Criteria Application Topical Report,” dated November 12, 1987; NRC (T.E. Murley to W.S. Wilgus) letter, “NRC Staff Review of Nuclear Steam Supply System Vendor Owners, Groups’ Application of the Commission’s Interim Policy Statement Criteria to Standard Technical Specification,” dated May 9, 1988, ADAMS Accession No. ML11264A057; TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 - Proposed Technical Specifications (TS),” dated August 27, 1992, ADAMS Accession No. ML073200281; and NRC letter, “Issuance of Facility Operating License No. NPF-90, Watts Bar Nuclear Plant, Unit 1(TAC M94025),” dated February 7, 1996, ADAMS Accession No. ML052930169.

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops shall be OPERABLE, and consist of either:

a. Any combination of RCS loops and residual heat removal (RHR) loops, and one loop shall be in operation, when the rod control system is not capable of rod withdrawal; or

b. Two RCS loops, and both loops shall be in operation, when the rod control system is capable of rod withdrawal.

--NOTES---

1. No RCP shall be started with any RCS cold leg temperature ≤ 350ºF unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures.

2. For the initial 7 hours after entry into MODE 3 from MODE 1 or MODE 2, two loops shall consist of:

   a. Two RCS loops with one loop in operation when the rod control system is not capable of rod withdrawal; or

   b. Two RCS loops with both loops in operation when the rod control system is capable of rod withdrawal.

3. Average reactor coolant temperature shall be maintained > 200ºF for the initial 7 hours after entry into MODE 3 from MODE 1 or MODE 2.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Only one RCS loop OPERABLE. <u>AND</u> Two RHR loops inoperable. <u>OR</u> Less than 7 hours since entry into MODE 3 from MODE 1 or MODE 2.	A.1	Initiate action to restore a second required RCS or RHR loop to OPERABLE status.	Immediately
B. One required RHR loop inoperable. <u>AND</u> No RCS loops OPERABLE.	B.1	Be in MODE 5.	24 hours
C. One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal.	C.1	Restore required RCS loop to operation.	1 hour
	<u>OR</u>		
	C.2	De-energize all control rod drive mechanisms (CRDMs).	1 hour
D. Required RCS or RHR loops inoperable. <u>OR</u> No required RCS or RHR loop in operation	D.1	De-energize all CRDMs.	Immediately
	<u>AND</u>		
	D.2	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u>		
	D.3	Initiate action to restore one required loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify two RCS loops are in operation when the rod control system is capable of rod withdrawal.	12 hours
SR 3.4.6.2	Verify one required RHR or RCS loop is in operation when the rod control system is not capable of rod withdrawal.	12 hours
SR 3.4.6.3	Verify SG secondary side water levels are greater than or equal to 32% narrow range for required RCS loops.	12 hours
SR 3.4.6.4	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

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

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

In MODE 4, with the reactor trip breakers open and the rods not capable of withdrawal, either RCPs or RHR loops can be used to provide forced circulation. The intent in this case is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent is to require that two paths be available to provide redundancy for decay heat removal.

In MODE 4, with the reactor trip breakers closed and the rods capable of withdrawal, two RCPs must be OPERABLE and in operation to provide forced circulation.

During a normal shutdown, decay heat removal is via the RCS loops until sometime after the unit has been cooled down to RHR entry conditions (T_{cold} < 350ºF). Therefore, as LCO 3.4.6 becomes Applicable (entry into MODE 4) the RCS loops are still OPERABLE. Transitioning decay heat removal to the RHR System will place high heat loads on the RHR System, Component Cooling System (CCS), and the Essential Raw Cooling Water System (ERCW). Residual and decay heat from the RCS is transferred to CCS via the RHR HX. Heat from the CCS is transferred to the ERCW System via the CCS HXs. The CCS and ERCW systems are common between the two operating units.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

In MODE 4, with the reactor trip breakers open and the rods not capable of withdrawal, RCS circulation is considered in determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 4 with RTBs in the closed position and Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, any combination of two RCS or RHR loops are required to be OPERABLE, but only one loop is required to be in operation to meet decay heat removal requirements, except during the initial seven hours after unit shutdown, when the decay and latent heat load may exceed the heat removal capability of one RHR loop in operation.

During the initial seven hours after reactor shutdown, the heat loads are at sufficiently high levels that the requirement of LCO 3.4.6 for one RHR loop in operation may not be sufficient to mitigate a design basis accident on Unit 1 and preclude a heatup of Unit 2.

To assure that there would be adequate heat removal capability under all postulated conditions during the initial seven hours after unit shutdown, reliance on heat removal via RCS loops is required. After a unit has been shutdown for greater than seven hours, a single RHR loop in operation provides adequate heat removal capability.

RCS Loops - MODE 4 have been identified in the NRC Policy Statement as important contributors to risk reduction.

(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE. In MODE 4 with the RTBs in the closed position and Rod Control System capable of rod withdrawal, two RCS loops must be OPERABLE and in operation. Two RCS loops are required to be in operation in MODE 4 with RTBs closed and Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

With the RTBs in the open position, or the CRDMs de-energized, the Rod Control System is not capable of rod withdrawal; therefore, only one loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. In this case, the LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature less than or equal to 350ºF. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 2 requires two RCS loops to be OPERABLE during the initial seven hours after entry into MODE 3 from MODE 1 or MODE 2 until decay heat and latent heat are within the capacity of the RHR System.

Note 3 precludes entry into MODE 5 during the initial seven hours after entry into MODE 3 from MODE 2 or MODE 1. This ensures that heat removal capability via RCS loops in retained until decay heat and latent heat are within the capacity of the RHR System.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG, which has the minimum water level specified in SR 3.4.6.3.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

(continued)

BASES (continued)

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

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 3.4.7, "RCS Loops - MODE 5, Loops Filled";

LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";

LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If only one RCS loop is OPERABLE and both RHR loops are inoperable, redundancy for heat removal is lost, and action must be initiated to restore a second RCS or RHR loop to OPERABLE status. If only one RCS loop is OPERABLE and it has been less than seven hours since the unit has entered MODE 3 from MODE 1 or MODE 2, redundancy has been lost and action must be initiated to restore a second RCS loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1

If one required RHR loop is OPERABLE and in operation and there are no RCS loops OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the plant must be brought to MODE 5 within 24 hours. Bringing the plant to MODE 5 is a conservative action with regard to decay heat removal. With only one required RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 (less than or equal to 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of

(continued)

ACTIONS

B.1 (continued)

24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

C.1 and C.2

If one required RCS loop is not in operation, and the RTBs are closed and Rod Control System capable of rod withdrawal, the Required Action is either to restore the required RCS loop to operation or to de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets. When the RTBs are in the closed position and Rod Control System capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the RTBs must be opened. The Completion Times of 1 hour to restore the required RCS loop to operation or de-energize all CRDMs is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2 and D.3

If no loop is OPERABLE or in operation, all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one required RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. Opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.2

This SR requires verification every 12 hours that one required RCS or RHR loop is in operation when the rod control system is not capable of rod withdrawal. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.3

SR 3.4.6.3 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is greater than or equal to 32% (value does not account for instrument error, Ref. 1). If the SG secondary side narrow range water level is less than 32%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12-hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.4

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. NRC Commission, "Interim Staff Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," Federal Register 52 FR 3788, dated February 6, 1987; Westinghouse Owners Group (R.A. Newton) letter to NRC Document Control Desk, "Westinghouse Owners Group MERITS Program Phase II, Task 5, Criteria Application Topical Report," dated November 12, 1987; NRC (T.E. Murley to W.S. Wilgus) letter, "NRC Staff Review of Nuclear Steam Supply System Vendor Owners, Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specification," dated May 9, 1988, ADAMS Accession No. ML11264A057; TVA letter, "Watts Bar Nuclear Plant (WBN) Unit 1 - Proposed Technical Specifications (TS)," dated August 27, 1992, ADAMS Accession No. ML073200281; and NRC letter, "Issuance of Facility Operating License No. NPF-90, Watts Bar Nuclear Plant, Unit 1(TAC M94025)," dated February 7, 1996, ADAMS Accession No. ML052930169.

(continued)

3.7 PLANT SYSTEMS

3.7.16 Component Cooling System (CCS) - Shutdown

LCO 3.7.16 Two CCS trains shall be OPERABLE with one pump powered from Train A and aligned to the Train A header, and two pumps powered from Train B and aligned to the Train B header.

APPLICABILITY: MODES 4 and 5.

---NOTE---
This LCO is not applicable more than 48 hours after entry into MODE 3 from MODE 1 or 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCS train inoperable in MODE 4. <u>AND</u> Complying with Required Actions to be in MODE 5.	A.1 Be in MODE 5.	24 hours
B. One CCS train inoperable in MODE 4 for reasons other than Condition A.	B.1 Verify two OPERABLE reactor coolant system (RCS) loops and one RCS loop in operation. <u>AND</u> B.2 Verify T_{avg} > 200ºF.	Once per 12 hours Once per 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two CCS trains inoperable in MODE 4.	C.1 ---------------NOTES-------------- 1. LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one CCS train is restored to an OPERABLE status. 2. Enter Conditions and Required Actions of LCO 3.4.6, “RCS Loops - MODE 4,” for residual heat removal (RHR) loops made inoperable by CCS. -- Initiate action to restore one CCS train to OPERABLE status.	 Immediately
D. One or more CCS train(s) inoperable in MODE 5.	D.1 ---------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," for RHR loops made inoperable by CCS. -- Initiate action to restore CCS train(s) to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify correct breaker alignment and indicated power available to the required pump(s) that is not in operation.	12 hours
SR 3.7.16.2	Verify two CCS pumps are aligned to CCS Train B.	12 hours

B 3.7 PLANT SYSTEMS

B 3.7.16 Component Cooling System (CCS) - Shutdown

BASES

BACKGROUND

The general description of the Component Cooling System (CCS) is provided in TS Bases 3.7.7, “Component Cooling System.” The CCS has a Unit 2 Train A header supplied by CCS Pump 2A-A cooled through CCS Heat Exchanger (HX) A. Unit 1 has a separate Train A header containing HX B supplied by CCS Pump 1A-A. The Train B header is shared by Unit 1 and Unit 2 and contains HX C. Flow through the Train B header is normally supplied by CCS Pump C-S. CCS Pump 1B-B can be aligned to supply the Train B header, but it is normally aligned to the Unit 1 Train A header. Similarly, CSS Pump 2B-B can supply cooling water to the Train B header, but is normally aligned to the Unit 2 Train A header. The following describes the functions and requirements within the first 48 hours after shut down, when the Residual Heat Removal (RHR) System is being used for residual and decay heat removal.

During a normal shutdown, decay heat removal is via the reactor coolant system (RCS) loops until sometime after the unit has been cooled down to RHR entry conditions (T_{cold} < 350ºF). Therefore, as LCO 3.7.16 becomes Applicable (entry into Mode 4) the RCS loops are still OPERABLE. Entry into MODES 4 and 5 can place high heat loads onto the RHR System, CCS and the Essential Raw Cooling Water System (ERCW) when shutdown cooling is established. Residual and decay heat from the Reactor Coolant System (RCS) is transferred to CCS via the RHR HX. Heat from the CCS is transferred to the ERCW System via the CCS HXs. The CCS and ERCW systems are common between the two operating units.

During the first 48 hours after reactor shutdown, the heat loads are at sufficiently high levels that the normal pump requirement of LCO 3.7.7 for one CCS pump on the Train B header may not be sufficient to support shut down cooling of Unit 2, concurrent with either a nearly simultaneous shutdown of Unit 2 or a design basis loss of coolant accident (LOCA) on Unit 2, with loss of offsite power and a single failure of Train A power to 6.9 kV Shutdown Boards 1A-A and 2A-A.

In either scenario, CCS Pump C-S would normally be the only pump supplying the Train B header and the Train B header would be supplying both the Unit 1 RHR Train B HX and the Unit 2 RHR Train B HX. During the Unit 1 LOCA scenario, the Unit 1 RHR Train B HX would be cooling the recirculating Emergency Core Cooling System (ECCS) water from the containment sump.

(continued)

BASES

BACKGROUND (continued)

To assure that there would be adequate CCS flow to both units' RHR Train B HXs, prior to placing RHR in service for Unit 2, either CCS Pump 1B-B or 2B-B would be aligned to the CCS Train B header.

With two CCS pumps on the Train B header, CCS will supply at least 5000 gpm to the Unit1 RHR Train B HX and 5000 gpm to the Unit 2 RHR Train B HX.

The alignment of either CCS Pump 1B-B or 2B-B to the CCS Train B header before entry into MODE 4 places both units in an alignment that supports LOCA heat removal requirements and allows the other unit to proceed to cold shutdown. Having the CCS pumps realigned while a unit being shut down with steam generators available for heat removal, precludes the need for manual action outside of the main control room to align CCS should a LOCA occur. If a LOCA occurs with the concurrent loss of the Train A 6.9 kV shutdown boards, CCS Pump 1B-B or 2B-B will be started from the main control room, if the pump is not already in operation. Both CCS pumps must be running before the RHR pump suction is transferred from the refueling water storage tank (RWST) to the containment sump to ensure adequate cooling is maintained. If a LOCA occurs, the C-S pump automatically starts on a safety injection (SI) actuation from either unit. The CCS pump control circuits are designed such that, if a pump is running and a loss of power occurs, the pump will be automatically reloaded on the DG. With this alignment, two CCS pumps will be available if a LOCA occurs on one unit when the other unit is being shut down.

Alternatively, the unit being shut down can remain on steam generator cooling for 48 hours before RHR is placed in service. If a LOCA occurred on the other unit, CCS would only be removing heat from one RHR HX. A single CCS pump and CCS HX provides the required heat removal capability.

After the unit has been shut down for greater than 48 hours, a single CCS pump on Train B provides adequate flow to both the Unit 1 and the Unit 2 RHR Train B HXs.

If the single failure were the loss of Train B power, the normal CCS alignment is acceptable, because CCS Pump 1A-A supplies the Unit 1 RHR Train A HX and CCS Pump 2A-A supplies the Unit 2 RHR Train A HX. CCS Pump 2A-A does not provide heat removal for Unit 1.

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 heat from the reactor via the RHR System. This may be

(continued)

BASES

BACKGROUND (continued)	during a normal or post accident cool down and shut down. The Unit 1 CCS Train A header is not used to support Unit 2 operation.

APPLICABLE SAFETY ANALYSES

The CCS functions to cool the unit from RHR entry conditions in MODE 4 (T_{cold} < 350°F), to MODE 5 (T_{cold} < 200°F), during normal operations. The time required to cool from 350°F to 200°F is a function of the number of CCS and RHR trains operating. One CCS train is sufficient to remove heat during subsequent operations with T_{cold} < 200°F. This assumes a maximum ERCW inlet temperature of 85°F occurring simultaneously with the maximum heat loads on the system.

The design basis of the CCS is for one CCS train to remove the post LOCA heat load from the containment sump during the recirculation phase, with a maximum CCS HX outlet temperature of 110°F (Ref. 2). The ECCS LOCA analysis and containment LOCA analysis each model the maximum and minimum performance of the CCS, respectively. The normal maximum HX outlet temperature of the CCS is 95°F, and, during unit cooldown to MODE 5 (T_{cold} < 200°F), a maximum HX outlet temperature of 110°F is assumed. The CCS design based on these values, bounds the post accident conditions such that the sump fluid will not increase in temperature after alignment of the RHR HXs during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the RCS by the ECCS pumps.

The CCS is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

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

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BASES

LCO

The CCS trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. During a unit shut down, one CCS train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two trains of CCS must be OPERABLE. At least one CCS train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

This LCO provides CCS train OPERABILITY requirements beyond the requirements of LCO 3.7.7 during the first 48 hours after reactor shut down, when the heat loads are at sufficiently high levels that the normal pump requirement of one CCS pump on the Train B header may not be sufficient to support shutdown cooling of Unit 2, concurrent with a nearly simultaneous shutdown of Unit 1 or a LOCA on Unit 1, a loss of offsite power, and single failure of Train A power to 6.9 kV Shutdown Boards 1A-A and 2A-A.

Because CCS Train B supports heat removal from Unit 1 and Unit 2, when Unit 2 has been shutdown ≤ 48 hours and the RHR System is relied on for heat removal, the following is required for CCS OPERABILITY:

a. Train A is OPERABLE when CCS Pump 2A-A is available and aligned to the CCS Train A header.

b. Train B is OPERABLE when two CCS pumps are available and aligned to the CCS Train B header using any combination of CCS Pumps 1B-B, 2B-B, and C-S.

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

Because Unit 2 is shutdown and on RHR cooling, no automatic actuations are required as a DBA on Unit 2, such as a LOCA, does not have to be mitigated.

APPLICABILITY

Prior to aligning the RHR System for RCS heat removal in MODE 4, an additional CCS pump must be powered from and aligned to the CCS Train B header to ensure adequate heat removal capability.

The Applicability is modified by a Note stating the LCO does not apply after the initial 48 hours after the unit enters MODE 3 from MODE 1 or MODE 2. Following extended operation in MODE 1, the heat loads are at sufficiently high levels that the normal pump requirement of LCO 3.7.7

(continued)

BASES

APPLICABILITY (continued)

for one CCS pump on the Train B header may not be sufficient to support shutdown cooling of Unit 2, concurrent with a near simultaneous shutdown of Unit 1 or a design basis LOCA on Unit 1, with loss of offsite power and a single failure of Train A power to 6.9 kV Shutdown Boards 1A-A and 2A-A. However, after the initial 48 hours following shutdown of the unit, the heat removal capability of both units is within the capabilities of the CCS without the need for an additional CCS pump aligned to the CCS Train B header.

ACTIONS

A.1

In MODE 4, if one CCS train is inoperable, and the unit is required to be placed in MODE 5 to comply with Required Actions, action must be taken to place the unit in MODE 5 within 24 hours. When the Required Actions of an LCO direct the unit to be placed in MODE 5, either a loss of safety function has occurred or the Required Action and Completion Time for restoring a safety-related component has not been met. Therefore, it is prudent to place the unit in a condition of lower energy with a lower potential for a postulated event. In this Condition, the remaining OPERABLE CCS train is adequate to perform the heat removal function. The 24 hour Completion Time is consistent with LCO 3.4.6, “RCS Loops - MODE 4,” Required Action B.1 for the Condition of one required RHR loop inoperable and no RCS loops OPERABLE.

B.1 and B.2

In MODE 4, if one CCS train is inoperable, and the unit is not required to be placed in MODE 5 to comply with Required Actions, actions are taken to verify LCO 3.4.6 is being met with two OPERABLE RCS loops with one loop in operation, and that the unit remains in MODE 4 (T_{avg} > 200ºF). These actions indicate the preference to maintain the unit in a condition with multiple methods of decay heat removal available, i.e, maintain the unit in MODE 4 with two RCS loops operable in addition to the remaining OPERABLE RHR loop. This action precludes entry into the LCO 3.4.6 Actions, as LCO 3.4.6 is met with two OPERABLE RCS loops and one RCS loop in operation. This Action is conservative to the Required Actions of LCO 3.4.6 when there are two OPERABLE RCS loops.

Maintaining the unit in MODE 4 with additional methods of decay heat removal available minimizes the likelihood of a situation where the decay heat and residual heat of the unit exceeds the capability of the available RHR loop resulting in the possibility of an unintentional MODE change. The Frequency of once per 12 hours ensures that the systems being relied on for heat removal are operating properly and are maintaining the unit in MODE 4. The 12 hour Frequency

(continued)

ACTIONS

B.1 and B.2 (continued)

is reasonable, considering the low probability of a change in system operation during this time period.

If the Required Actions and Completion Times of Condition B are not met, no actions are specified. Therefore, LCO 3.0.3 applies, requiring the unit to be placed in MODE 5 in 37 hours. With one CCS train inoperable and Required Actions require the unit to be placed in MODE 5, Condition A applies, requiring the unit to be placed in MODE 5 in 24 hours. This Action is consistent with the Required Actions of LCO 3.4.6 Condition B (no OPERABLE RCS loops and one inoperable RHR loop).

C.1

In MODE 4, if two CCS trains are inoperable, immediate action must be taken to restore one of the CCS trains to an OPERABLE status, as no CCS train is available to support the heat removal function. Required Action C.1 is consistent with LCO 3.4.6, “RCS Loops - MODE 4,” Required Action D.1 for the Condition of required RCS or RHR loops inoperable and no required RCS or RHR loop in operation.

Required Action C.1 is modified by two Notes. Note 1 indicates that all required MODE changes or power reductions are suspended until one CCS train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition. Note 2 indicates that the applicable Conditions and Required Actions of LCO 3.4.6 be entered for RHR loops made inoperable by the inoperable CCS trains. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

D.1

Required Action D.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.7, “RCS Loops - MODE 5, Loops Filled,” be entered for RHR loops made inoperable by one or more inoperable CCS train(s). This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

In MODE 5, if one or more CCS train(s) is inoperable, action must be initiated immediately to restore the CCS train(s) to an OPERABLE status to restore heat removal paths. The immediate Completion Time reflects the importance of maintaining the capability of heat removal.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

Verification that each required CCS pump that is not in operation is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain heat removal. Verification is performed by verifying proper breaker alignment and power available to the CCS pump(s). The 12 hour Frequency is based on engineering judgment.

SR 3.7.16.2

This SR verifies that two of the three CCS pumps that are powered from Train B are aligned to the Train B header. Verification of the correct physical alignment assures that adequate CCS flow can be provided to both the Unit 1 and Unit 2 RHR Train B HXs, if required. The 12 hour Frequency is based on engineering judgment, is consistent with procedural controls governing valve alignment, and ensures correct valve positions.

REFERENCES

1. Watts Bar FSAR, Section 9.2.2, “Component Cooling System.”

2. Watts Bar Component Cooling System Description, 2-N3-70-4002.

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3.7 PLANT SYSTEMS

3.7.17 Essential Raw Cooling Water (ERCW) System - Shutdown

LCO 3.7.17 Two ERCW trains shall be OPERABLE as follows:

a. Three ERCW pumps aligned to Train A, including two pumps capable of being powered from 6.9 kV Shutdown Board 2A-A, and

b. Three ERCW pumps aligned to Train B, including two pumps capable of being powered from 6.9 kV Shutdown Board 2B-B.

APPLICABILITY: MODES 4 and 5.

---NOTE---

This LCO is not applicable more than 48 hours after entry into MODE 3 from MODE 1 or 2.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One ERCW train inoperable in MODE 4. <u>AND</u> Complying with Required Actions to be in MODE 5.	A.1	Be in MODE 5.	24 hours
B.	One ERCW train inoperable in MODE 4 for reasons other than Condition A.	B.1	Verify two OPERABLE reactor coolant system (RCS) loops and one RCS loop in operation.	Once per 12 hours
		<u>AND</u>		
		B.2	Verify T_{avg} > 200ºF.	Once per 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two ERCW trains inoperable in MODE 4.	C.1 ---------------NOTES-------------- 1. LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one ERCW train is restored to an OPERABLE status. 2. Enter Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal (RHR) loops made inoperable by ERCW. -- Initiate action to restore one ERCW train to OPERABLE status.	 Immediately
D. One or more ERCW train(s) inoperable in MODE 5.	D.1 ---------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," for RHR loops made inoperable by ERCW. -- Initiate action to restore ERCW train(s) to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify correct breaker alignment and indicated power available to the required pump(s) that is not in operation.	12 hours

B 3.7 PLANT SYSTEMS

B 3.7.17 Essential Raw Cooling Water (ERCW) System

BASES

BACKGROUND

The general description of ERCW is provided in TS Bases 3.7.8, "Essential Raw Cooling Water (ERCW) System." The descriptions of Applicable Safety Analyses, LCOs, Applicability, ACTIONS and Surveillance Requirements for applicable MODES are also described in TS Bases 3.7.8. The following discussion applies to the specific Applicability in TS 3.7.17 during the first 48 hours after shut down when the Residual Heat Removal (RHR) System is being used for residual and decay heat removal. The ERCW System 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, and a normal shutdown, the ERCW System also provides this function for various safety related and non-safety related components. The major post-accident heat load on the ERCW System is the Component Cooling System (CCS) heat exchangers (HXs), which are used to cool RHR and the containment spray HXs. The major heat load on the ERCW System when a unit is shut down on RHR is the CCS HX associated with the train(s) of RHR in service.

During a normal shutdown, decay heat removal is via the reactor coolant system (RCS) loops until sometime after the unit has been cooled down to RHR entry conditions (T_{cold} < 350ºF). Therefore, as LCO 3.7.17 becomes Applicable (entry into Mode 4) the RCS loops are still OPERABLE. After the RHR System is aligned as the principle method of decay heat removal, the heat loads on the ERCW System are increased. Normally, two ERCW pumps are sufficient to handle the cooling needs for maintaining one unit in normal operation while mitigating a DBA on the other unit. However, in the unlikely event of a loss of coolant accident (LOCA) on Unit 2 with a concurrent loss of offsite power and a single failure that results in the loss of both Train A or both Train B 6.9 kV shutdown boards while Unit 1 is on RHR shutdown cooling and has been shutdown for less than 48 hours, three ERCW pumps may be required.

This LCO controls the availability of ERCW pumps necessary to support mitigation of a LOCA on Unit 2 when Unit 1 has been shut down for ≤ 48 hours and is utilizing RHR for heat removal.

Additional information about the design and operation of the ERCW System, along with a list of the components served, is presented in the FSAR, Section 9.2.1 (Ref. 1).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The design basis of the ERCW System is for one ERCW train, in conjunction with the CCS and a 100% capacity Containment Spray System and RHR, to remove core decay heat following a design basis LOCA as discussed in the FSAR, Section 9.2.1 (Ref. 1). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the Emergency Core Cooling System (ECCS) pumps. The ERCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The ERCW System, in conjunction with the CCS, also cools the unit, as discussed in the FSAR, Section 5.5.7 (Ref. 2) from RHR entry conditions to MODE 5 during normal and post accident operations. The time required to enter MODE 5 is a function of the number of CCS and RHR System trains that are operating. One ERCW train is sufficient to remove heat during subsequent operations in MODES 5 and 6. This assumes a maximum ERCW inlet temperature of 85ºF occurring simultaneously with maximum heat loads on the system. In the first 48 hours after the shutdown of Unit 1 assuming a DBA LOCA on Unit 2 with the loss of offsite power and the concurrent loss of two 6.9 kV shutdown boards on the same power train as a single failure. Three ERCW pumps are required to provide the heat removal capacity assumed in the safety analysis for Unit 2 while continuing the cooldown of Unit 1.

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

LCO

This LCO provides ERCW train OPERABILITY requirements beyond the requirements of LCO 3.7.8 during the first 48 hours after reactor shutdown, when the heat loads are at sufficiently high levels that the normal pump requirement of two ERCW pumps on one train may not be sufficient to support shutdown cooling of Unit 1, concurrent with a LOCA on Unit 2, an assumed loss of offsite power, and a single failure that affects both 6.9 kV shutdown boards in one power train.

Two ERCW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to support a cooldown to MODE 5.

An ERCW train is considered OPERABLE during the first 48 hours after shutdown when:

a. Two pumps per train, aligned to separate shutdown boards, are OPERABLE; and

(continued)

LCO (continued)

b. One additional Train A pump and one additional Train B pump are capable of being aligned to their respective Unit 1 6.9 kV shutdown board (1A-A and 1B-B) and manually placed in service.

APPLICABILITY

Prior to aligning the RHR System for RCS heat removal in MODE 4, one additional ERCW pump must be capable of being powered by its respective Unit 1 6.9 kV shutdown board (1A-A and 1B-B) and manually placed in service to ensure adequate heat removal capability.

The Applicability is modified by a Note stating the LCO does not apply after the initial 48 hours after the unit enters MODE 3 from MODE 1 or MODE 2. Following extended operation in MODE 1, the heat loads are at sufficiently high levels that the normal pump requirement of LCO 3.7.8 for two ERCW pumps may not be sufficient to support shutdown cooling of Unit 1, concurrent with a design basis LOCA on Unit 2 with loss of offsite power and a single failure of Train A power to 6.9 kV Shutdown Boards 1A-A and 2A-A. However, after the initial 48 hours following shutdown of the unit, the heat removal capability of both units is within the capabilities of the ERCW System without the need for an additional ERCW pump in each train.

ACTIONS

A.1

In MODE 4, if one ERCW train is inoperable, and the unit is required to be placed in MODE 5 to comply with Required Actions, action must be taken to place the unit in MODE 5 within 24 hours. When the Required Actions of an LCO direct the unit to be placed in MODE 5, either a loss of safety function has occurred or the Required Action and Completion Time for restoring a safety-related component has not been met. Therefore, it is prudent to place the unit in a condition of lower energy with a lower potential for a postulated event. In this Condition, the remaining OPERABLE ERCW train is adequate to perform the heat removal function. The 24 hour Completion Time is consistent with LCO 3.4.6, “RCS Loops - MODE 4,” Required Action B.1 for the Condition of one required RHR loop inoperable and no RCS loops OPERABLE.

B.1 and B.2

In MODE 4, if one ERCW train is inoperable, and the unit is not required to be placed in MODE 5 to comply with Required Actions, actions are taken to verify LCO 3.4.6 is being met with two OPERABLE RCS loops with one loop in operation, and that the unit remains in MODE 4 (T_{avg} > 200ºF). These actions indicate the preference to maintain the unit

(continued)

ACTIONS

B.1 (continued)

in a condition with multiple methods of decay heat removal available, i.e., maintain the unit in MODE 4 with two RCS loops operable in addition to the remaining OPERABLE RHR loop. This action precludes entry into the LCO 3.4.6 Actions, as LCO 3.4.6 is met with two OPERABLE RCS loops and one RCS loop in operation. This Action is conservative to the Required Actions of LCO 3.4.6 when there are two OPERABLE RCS loops.

Maintaining the unit in MODE 4 with additional methods of decay heat removal available minimizes the likelihood of a situation where the decay heat and residual heat of the unit exceeds the capability of the available RHR loop resulting in the possibility of an unintentional MODE change. The Frequency of once per 12 hours ensures that the systems being relied on for heat removal are operating properly and are maintaining the unit in MODE 4. The 12 hour Frequency is reasonable, considering the low probability of a change in system operation during this time period.

If the Required Actions and Completion Times of Condition B are not met, no actions are specified. Therefore, LCO 3.0.3 applies, requiring the unit to be placed in MODE 5 in 37 hours. With one ERCW train inoperable and Required Actions require the unit to be placed in MODE 5, Condition A applies, requiring the unit to be placed in MODE 5 in 24 hours. This Action is consistent with the Required Actions of LCO 3.4.6 Condition B (no OPERABLE RCS loops and one inoperable RHR loop).

Although LCO 3.7.17 provides requirements in addition to those of LCO 3.7.8, the additional requirements of LCO 3.7.17 are not required for DG OPERABILITY. There is sufficient flow to the DGs from ERCW without a third ERCW in each train to support DG OPERABILITY. Although the requirement of LCO 3.7.17 may not be met (i.e., a third pump capable of being aligned to each ERCW Train) the requirement of LCO 3.7.8 is still met. If the requirement of LCO 3.7.8 is not met, the Actions of LCO 3.7.8 include the requirement to enter the Conditions and Required Actions of LCO 3.8.1 for DGs made inoperable by ERCW.

C.1

In MODE 4, if two ERCW trains are inoperable, immediate action must be taken to restore one of the ERCW trains to an OPERABLE status, as no ERCW train is available to support the heat removal function. Required Action C.1 is consistent with LCO 3.4.6, “RCS Loops - MODE 4,”

(continued)

ACTIONS

C.1 (continued)

Required Action D.1 for the Condition of required RCS or RHR loops inoperable and no RCS or RHR loop in operation.

Required Action C.1 is modified by two Notes. Note 1 indicates that all required MODE changes or power reductions are suspended until one ERCW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition. Note 2 indicates that the applicable Conditions and Required Actions of LCO 3.4.6 be entered for RHR loops made inoperable by the inoperable ERCW trains. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

D.1

Required Action D.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.7, “RCS Loops - MODE 5, Loops Filled,” be entered for RHR loops made inoperable by one or more inoperable ERCW train(s). This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

In MODE 5, if one or more ERCW train(s) is inoperable, action must be initiated immediately to restore the ERCW train(s) to an OPERABLE status to restore heat removal paths. The immediate Completion Time reflects the importance of maintaining the capability of heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

Verifying the availability of the ERCW pumps provides assurance that adequate ERCW flow is provided for heat removal. Verification that each required ERCW pump that is not in operation is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal. Verification is performed by verifying proper breaker alignment and power available to the ERCW pump(s). The ERCW pump Interlock Bypass Switches do not need to be in ‘Bypass’ in order to meet this SR. The associated ERCW pump Interlock Bypass Switch is positioned by procedure when the third ERCW pump in the respective train is required to be started. The 12 hour Frequency is based on engineering judgment.

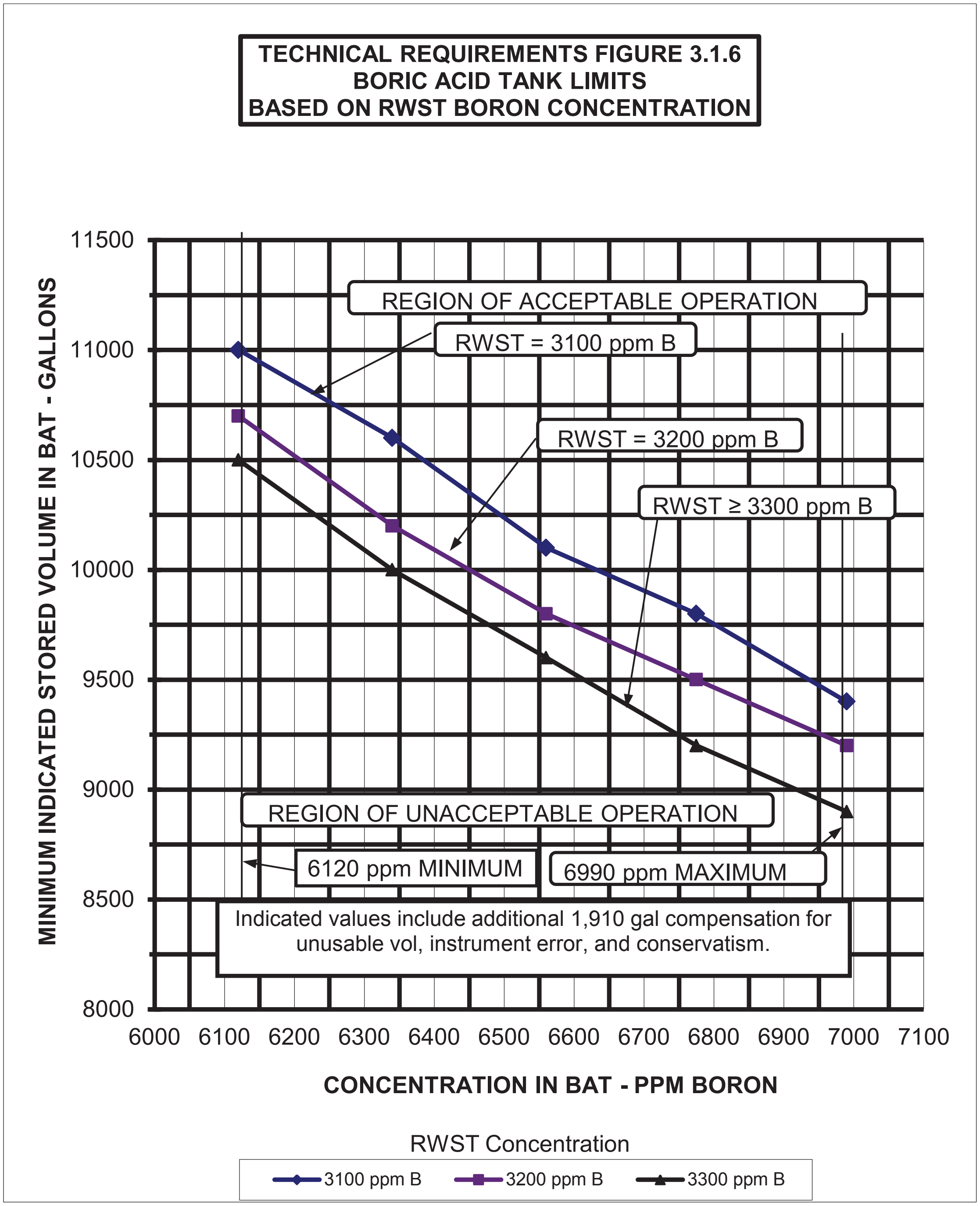
REFERENCES

1. Watts Bar FSAR, Section 9.2.1, "Essential Raw Cooling Water."
2. Watts Bar FSAR, Section 5.5.7, "Residual Heat Removal System."

Attachment 4
Watts Bar Nuclear Plant Unit 2

Technical Requirements Manual Bases

Section 3.1.6, Boric Acid Tank Limits
Clean


TECHNICAL REQUIREMENTS FIGURE 3.1.6
BORIC ACID TANK LIMITS
BASED ON RWST BORON CONCENTRATION
REGION OF ACCEPTABLE OPERATION
RWST = 3100 ppm B
RWST = 3200 ppm B
RWST ≥ 3300 ppm B
REGION OF UNACCEPTABLE OPERATION
6120 ppm MINIMUM
6990 ppm MAXIMUM
Indicated values include additional 1,910 gal compensation for unusable vol, instrument error, and conservatism.
MINIMUM INDICATED STORED VOLUME IN BAT - GALLONS
11500
11000
10500
10000
9500
9000
8500
8000
6000
6100
6200
6300
6400
6500
6600
6700
6800
6900
7000
7100
CONCENTRATION IN BAT - PPM BORON
RWST Concentration
3100 ppm B
3200 ppm B
3300 ppm B