Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

February 16, 2011 10 CFR 50.36

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

Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: Watts Bar Nuclear Plant (WBN) Unit 2 – Supplemental Information For Review of Developmental Revision A of the Unit 2 Technical Specifications (TS) and Technical Specifications Bases (TS Bases)

TVA's letter to NRC dated March 4, 2009 (Reference 1), included Developmental Revision A of the Unit 2 TS and TS Bases. Enclosure 2 of that letter noted that the proposed WBN Unit 2 TS and TS Bases were developed by marking up Revision 0 of NUREG-1431, "Standard Technical Specifications Westinghouse Plants," to indicate the WBN Unit 1 TS and TS Bases approved (as appropriate) through Amendment 70 and Revision 91, respectively.

Recently, an NRC staff member noted that the current licensing basis for Unit 1 was to be used as the reference basis for the review and licensing of Unit 2. Based on this, the staff member noted that the Developmental A versions of the TS and TS Bases should have been based on the Unit 1 TS amendment and TS Bases revision that were current when the first WBN Regulatory Framework letter was issued for Unit 2. As a result, the staff requested that a markup of these versions of the TS and TS Bases (i.e., when the framework letter was submitted) be provided that reflects the Unit 1 TS amendment and TS Bases revision that were current when Developmental Revision A of the Unit 2 TS and TS Bases was submitted.

To accomplish this activity, this letter provides the following:

- Enclosure 1 describes the process used to produce the requested markups of the TS and TS Bases. Additionally, the review matrix used for the markups is discussed.
- Enclosure 2 includes the TS Review Matrix and the associated marked-up TS pages.
- Enclosure 3 includes the TS Bases Review Matrix and the associated marked-up TS Bases pages.

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During the markup activity described in Enclosure 1, two new discrepancies in Reference 1 were identified. Enclosure 4 summarizes the two new discrepancies and the two discrepancies that were previously fixed in Reference 3 (details are provided in the applicable review matrix) and discusses the corrective action for each discrepancy.

Enclosure 5 provides the list of commitments made in this letter. If you have any questions, please contact Bill Crouch at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16th day of February, 2011.

Respectfully,

Marie Gillman
Acting Watts Bar Unit 2 Vice President

Enclosures:

1. Process for Marking Up the Technical Specifications (TS) and TS Bases Pages
2. Unit 2 TS Review Matrix: Unit 1 Amendments 67 through 70 With Associated Markups
3. Unit 2 TS Bases Review Matrix: Unit 1 Revisions 86 through 91 With Associated Markups
4. Discrepancies Identified During the Process of Marking Up the TS and TS Bases
5. Commitments

References:

1. TVA to NRC letter dated March 4, 2009, "Watts Bar Nuclear Plant (WBN) Unit 2 - Operating License Application Update" (ADAMS Accession No. ML090700378) [Developmental Revision A]

2. TVA to NRC letter dated January 29, 2008, "Watts Bar Nuclear Plant (WBN) - Unit 2 - Regulatory Framework for the Completion of Construction and Licensing Activities for Unit 2" (ADAMS Accession No. ML080320443)

3. TVA to NRC letter dated February 2, 2010, "Watts Bar Nuclear Plant (WBN) - Unit 2 - Developmental Revision B of the Technical Specifications (TS), TS Bases, Technical Requirements Manual (TRM), TRM Bases; and Pressure and Temperature Limits Report (PTLR)" (ADAMS Accession No. ML100550326)

4. TVA to NRC letter dated August 16, 2010, "Watts Bar Nuclear Plant (WBN) - Unit 2 - Change to Developmental TS Section 4.2.2, 'Control Rod Assemblies'" (ADAMS Accession No. ML102290075) [Developmental Revision C]

5. TVA to NRC letter dated October 12, 2010, "Watts Bar Nuclear Plant (WBN) - Unit 2 - Change to Developmental Technical Specification (TS) Sections 3.6.11, 'Ice Bed,' and 3.1.8, 'Rod Position Indication'" (ADAMS Accession No. ML1028505200) [Developmental Revision D]

6. TVA to NRC letter dated January 27, 2011, "Watts Bar Nuclear Plant (WBN) - Unit 2 - Change to Developmental Technical Specification (TS) Section 3.1.8, 'Rod Position Indication'" (ADAMS Accession No. ML110270108) [Developmental Revision E]

cc (Enclosures):

U. S. Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
Atlanta, Georgia 30303-1257

NRC Resident Inspector Unit 2
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

ENCLOSURE 1

Process for Marking Up the Technical Specifications (TS) and TS Bases Pages

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See Page 3 of the cover letter for references.

Developmental Revision A of the Unit 2 TS and TS Bases (Reference 1) were based on Revision 0 of NUREG-1431, "Standard Technical Specifications Westinghouse Plants." Developmental Revision A was constructed by starting with NUREG-1431 and then overlaying the Unit 1 TS amendments and TS Bases revisions made from Amendment 0 / Revision 0 through Amendment 70 / Revision 91, respectively.

The NRC staff reviewer has requested that an alternate markup be provided which is based on the versions of the Unit 1 TS and TS Bases which were current at the time of the first WBN Unit 2 regulatory framework letter (Reference 2). At that time, the Unit 1 TS and TS Bases were at Amendment 67 and Revision 86, respectively. Therefore, the attached markups depict the applicable changes from the date of Reference 2 to the date of Reference 1 (i.e., January 29, 2008, to March 4, 2009). Thus, the markups herein address Unit 1 TS Amendments 68 through 70 and TS Bases Revisions 87 through 91. Unit 1 TS Amendment 69 was found to be a temporary change that is not needed for Unit 2; thus, it is not included in the markup provided in Enclosure 2. Additionally, as noted below, some of the Unit 1 TS amendments and TS Bases revisions prior to Amendment 67 and Revision 86, respectively, will not apply to Unit 2 (e.g., Unit 2 will not utilize a tritium producing core).

Changes to the Unit 2 TS and TS Bases made after the date of Reference 1 are addressed by References 3 through 6.

The markups contained in this letter result in basically the same TS and TS Bases as provided in Reference 1 except for the discrepancies noted in Enclosure 4.

The requested markups provided in Enclosures 2 and 3 involve only the affected pages that resulted from the process described below:

<u>For the TS</u>:

- Marked up the applicable Unit 1 TS pages to show the changes that resulted from Unit 1 TS Amendments 68 and 70 (i.e., the amendments incorporated between the submittal of References 1 and 2).
- Designated the Unit 1 TS amendments that were not incorporated on Unit 2. Where appropriate, provided a markup to show how the TS looked prior to incorporation of the amendment that was not incorporated.
- Provided a review matrix for the TS. This matrix indicates, by TS section, which amendments were incorporated and which amendments were not incorporated. This matrix also delineates when / why a markup is not provided.

ENCLOSURE 1

Process for Marking Up the Technical Specifications (TS) and TS Bases Pages

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

For the TS Bases:

- Marked up the applicable Unit 1 TS Bases pages to show the changes that resulted from Unit 1 TS Bases Revisions 87, 88, 89, and 90 (i.e., the revisions incorporated between the submittal of References 1 and 2).
- Noted the Unit 1 TS Bases revisions that were not incorporated on Unit 2. Where appropriate, provided a markup to show how the TS Bases looked prior to incorporation of the revision that was not incorporated.
- Provided a review matrix for the TS Bases. This matrix indicates, by TS Bases section, which revisions were incorporated and which revisions were not incorporated. This matrix also delineates when / why a markup is not provided.

Key to Abbreviations Use on Review Matrices

- TS — Technical Specifications
- TS Bases — Technical Specifications Bases
- AXX — Technical Specifications Amendment XX
- RXX — Technical Specifications Bases Revision XX
- AXX (RXX) — Technical Specifications Amendment XX (Technical Specifications Bases Revision XX)

ENCLOSURE 2

Unit 2 TS Review Matrix: Unit 1 Amendments 67 through 70 With Associated Markups

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

UNIT 2 TS REVIEW MATRIX: UNIT 1 AMENDMENTS 67 THROUGH 70

TS SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS 1.1 Definitions	A31	A31 amended the Unit 1 TS to approve Power Uprate using Leading Edge Flow Meter (LEFM); this incorporated part of TSTF-51, R2. This change will NOT be incorporated into the Unit 2 TS at this time. NRC approved A31 via letter dated 01/19/2001.
TS 2.1 SLs	A31	A31 amended the Unit 1 TS to approve Power Uprate using Leading Edge Flow Meter (LEFM) for Unit 1. This change will NOT be incorporated into the Unit 2 TS at this time. NRC approved A31 via letter dated 01/19/2001.
TS 3.3.1 Reactor Trip System (RTS) Instrumentation	A47 ------ A68	A47 amended the Unit 1 TS to allow an alternate method for the measurement of RCS total flow rate via measurement of the RCS elbow tap differential pressures. Developmental Revision A for the Unit 2 TS inappropriately stated, "The changes will be applied to Unit 2." NRC approved A47 via letter dated 10/03/2003. ********** DISCREPANCY: The changes per A47 were inappropriately incorporated into Developmental Revision A of the Unit 2 TS. Developmental Revision B of the Unit 2 TS deleted the changes incorporated into the Unit 2 TS per A47; thus, an additional markup is NOT being provided. ---------- A68 amended the Unit 1 TS to allow relaxations of various logic completion times, bypass test times, allowable outage times, and surveillance testing intervals previously reviewed and approved by NRC under WCAP-14333-P-A, "Probabilistic Risk Analysis of RPS and ESFAS Test Times and Completion Times," and WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times." A68 also incorporated TSTF-169, "Deletion of Condition 3.3.1.N," and TSTF-311, "Revision of Surveillance Frequency for TADOT on Turbine Trip Functional Unit." The changes per A68 were incorporated herein. NRC approved A68 via letter dated 06/30/2008.

TS SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation	A23 ------ A68	A23 amended the Unit 1 TS for a ONE TIME exception to allow for relief from response time testing 1-FSV-47-027. This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS. NRC approved A23 via letter dated 03/22/2000. ---------- A68 amended the Unit 1 TS to allow relaxations of various logic completion times, bypass test times, allowable outage times, and surveillance testing intervals previously reviewed and approved by NRC under WCAP-14333-P-A, "Probabilistic Risk Analysis of RPS and ESFAS Test Times and Completion Times," and WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times." A68 also incorporated TSTF-169, "Deletion of Condition 3.3.1.N," and TSTF-311, "Revision of Surveillance Frequency for TADOT on Turbine Trip Functional Unit." The changes per A68 were incorporated herein. NRC approved A68 via letter dated 06/30/2008.
TS 3.3.4 Remote Shutdown System	A53	A53 amended the Unit 1 TS to allow a ONE TIME change to Function 4a of TS Table 3.3.4-1 (allows the Loop 4 RCS hot leg temperature indicator in the Auxiliary Control Room to be inoperable for the remainder of Cycle 6). This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS. NRC approved A53 via letter dated 11/19/2004.
TS 3.3.6 Containment Vent Isolation Instrumentation	A68	A68 amended the Unit 1 TS to allow relaxations of various logic completion times, bypass test times, allowable outage times, and surveillance testing intervals previously reviewed and approved by NRC under WCAP-14333-P-A, "Probabilistic Risk Analysis of RPS and ESFAS Test Times and Completion Times," and WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times." A68 also incorporated TSTF-169, "Deletion of Condition 3.3.1.N," and TSTF-311, "Revision of Surveillance Frequency for TADOT on Turbine Trip Functional Unit." The changes per A68 were incorporated herein. NRC approved A68 via letter dated 06/30/2008.
TS 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	A47	A47 amended the Unit 1 TS to allow an alternate method for the measurement of RCS total flow rate via measurement of the RCS elbow tap differential pressures. Developmental Revision A for the Unit 2 TS inappropriately stated, "The changes will be applied to Unit 2." NRC approved A47 via letter dated 10/03/2003. ********** DISCREPANCY: The changes per A47 were inappropriately incorporated into Developmental Revision A of the Unit 2 TS. Developmental Revision B of the Unit 2 TS deleted the changes incorporated per A47; thus, an additional markup is NOT being provided

TS SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS 3.4.5 RCS Loops - MODE 3	A61	A61 amended the Unit 1 TS to authorize change in steam generator (SG) level requirement from greater than or equal to 6% to greater than or 32% following SG replacement. This change will not be incorporated into the Unit 2 TS at this time since Unit 2 will utilize the original steam generators. NRC approved A61 via letter dated 05/05/2006.
TS 3.4.6 RCS Loops - MODE 4	A61	A61 amended the Unit 1 TS to authorize change in steam generator (SG) level requirement from greater than or equal to 6% to greater than or 32% following SG replacement. This change will not be incorporated into the Unit 2 TS at this time since Unit 2 will utilize the original steam generators. NRC approved A61 via letter dated 05/05/2006.
TS 3.4.7 RCS Loops - MODE 5, Loops Filled	A61	A61 amended the Unit 1 TS to authorize change in steam generator (SG) level requirement from greater than or equal to 6% to greater than or 32% following SG replacement. This change will not be incorporated into the Unit 2 TS at this time since Unit 2 will utilize the original steam generators. NRC approved A61 via letter dated 05/05/2006.
TS 3.4.12 Cold Overpressure Mitigation System (COMS)	A14	A14 amended the Unit 1 TS to allow up to 4 hours to make the residual heat removal suction relief valve available as a cold overpressure mitigation system (COMS) relief path. The changes implemented by A14 were superseded entirely by Unit 1 TS A55. Thus, changes per A14 will NOT be incorporated into the Unit 2 TS, and a markup is NOT provided. NRC approved A14 via letter dated 11/10/1998.
TS 3.4.13 RCS Operational LEAKAGE	A38	A38 amended the Unit 1 TS to incorporate voltage-based Alternate Repair Criteria for steam generator tubes. This change was NOT incorporated into the Unit 2 TS. Changes to TS 3.4.13 per A38 were revised in entirety by A65; thus, no markup is included. NRC approved A38 via letter dated 02/26/2002.

TS SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS 3.5.1 Accumulators	A40 ------ A48 ------ A67	A40 amended the Unit 1 TS to allow Watts Bar to irradiate up to 2304 Tritium Producing Burnable Absorber Rods (TPBARs) in the reactor core each fuel cycle. TPBARS will NOT be used on Unit 2. NRC approved A40 via letter dated 09/23/2002. ********** A67 was the current resolution of the boron concentration requirements of this issue for Unit 1. Since the Unit 2 TS markup is based on A67, no markup is required for either A40 or A48. ---------- A48 amended the Unit 1 TS to revise the boron concentration requirements and limit the number of Tritium Producing Burnable Absorber Rods (TPBARs) that could be loaded and irradiated in the core to a corresponding value. TPBARS will NOT be used on Unit 2. NRC approved A48 via letter dated 10/08/2003. ********** A67 was the current resolution of the boron concentration requirements of this issue for Unit 1. Since the Unit 2 TS markup is based on A67, no markup is required for either A40 or A48. ---------- A67 amended the Unit 1 TS to revise the maximum number of TPBARs that can be irradiated in the Unit 1 reactor core to 400. TPBARS will NOT be used on Unit 2; however, the boron concentration changes per this amendment were incorporated into the Unit 2 TS via Developmental Revision A of the Unit 2 TS. NRC approved A67 via letter dated 01/18/2008.
TS 3.5.2 ECCS - Operating	A43	A43 amended the Unit 1 TS to revise, for ONE TIME ONLY, a portion of SR 3.5.2.3. This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS. NRC approved A43 via letter dated 05/01/2003.

TS SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS 3.5.4 Refueling Water Storage Tank (RWST)	A40 ------ A48 ------ A67	A40 amended the Unit 1 TS to allow Watts Bar to irradiate up to 2304 Tritium Producing Burnable Absorber Rods (TPBARs) in the reactor core each fuel cycle. TPBARS will NOT be used on Unit 2. NRC approved A40 via letter dated 09/23/2002. ********** A67 was the current resolution of the boron concentration requirements of this issue for Unit 1. Since the Unit 2 TS markup is based on A67, no markup is required for either A40 or A48. ---------- A48 amended the Unit 1 TS to revise the boron concentration requirements and limit the number of Tritium Producing Burnable Absorber Rods (TPBARs) that could be loaded and irradiated in the core to a corresponding value. TPBARS will NOT be used on Unit 2. NRC approved A48 via letter dated 10/08/2003. ********** A67 was the current resolution of the boron concentration requirements of this issue for Unit 1. Since the Unit 2 TS markup is based on A67, no markup is required for either A40 or A48. ---------- A67 amended the Unit 1 TS to revise the maximum number of TPBARs that can be irradiated in the Unit 1 reactor core to 400. TPBARS will NOT be used on Unit 2; however, the boron concentration changes per this amendment were incorporated into Unit 2 via Developmental Revision A of the Unit 2 TS. NRC approved A67 via letter dated 01/18/2008.
TS 3.6.4 Containment Pressure	A59	A59 amended the Unit 1 TS to support steam generator replacement by allowing TEMPORARY use of penetrations in Shield Building Dome during Modes 1-4. This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS. NRC approved A59 via letter dated 01/06/2006.
TS 3.6.8 Hydrogen Mitigation System (HMS)	A10	A10 amended the Unit 1 TS for a ONE TIME EXCEPTION to allow certain hydrogen igniters to be inoperable for a LIMITED TIME PERIOD. This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS. NRC approved A10 via letter dated 06/09/1998.
TS 3.6.11 Ice Bed	A62	A62 amended the Unit 1 TS to support steam generator replacement - increased minimum ice weight and total weight of stored ice. Development Revision A to the Unit 2 TS did not include the change in minimum ice weight and total weight of stored ice; however, TVA later decided to include this conservative change in order to maintain unit fidelity. Developmental Revision D to the Unit 2 TS and TS Bases (submitted to the NRC on 10/12/2010) revised the ice weights to be consistent with Unit 2.

TS SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS 3.6.12 Ice Condenser Doors	A03	A03 amended the Unit 1 TS to allow a ONE TIME EXTENSION of the 3 month surveillance requirement for the ice condenser lower inlet doors. This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS. NRC approved A03 via letter date 09/09/1996.
TS 3.6.15 Shield Building	A59	A59 amended the Unit 1 TS to support steam generator replacement by allowing TEMPORARY use of penetrations in Shield Building Dome during Modes 1-4. This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS. NRC approved A59 via letter dated 01/06/2006.
TS 3.7.1 Main Steam Safety Valves (MSSVs)	A31	A31 amended the Unit 1 TS to approve Power Uprate using Leading Edge Flow Meter (LEFM) for Unit 1. This change will NOT be implemented on Unit 2 at this time. NRC approved A31 via letter dated 01/19/2001. ********** DISCREPANCY: Contrary to this statement, the "58%" value in REQUIRED ACTION A.1 should have remained at "59%." The "58%" value in REQUIRED ACTION A.1 will be corrected to "59%" via a future Developmental Revision to the Unit 2 TS.
TS 3.7.8 Essential Raw Cooling Water (ERCW) System	A69	A69 amended the Unit 1 TS as a ONE TIME CHANGE to address ERCW pumps A-1 and B-1 being inoperable. This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS. NRC approved A69 via letter dated 07/24/2008.
TS 3.7.10 Control Room Emergency Ventilation System (CREVS)	A70	A70 amended the Unit 1 TS to adopt TSTF Change Traveler TSTF-448, R3: Control Room Envelope Habitability. The changes per A70 were incorporated herein. NRC approved A70 via letter dated 10/08/2008.

TS SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS 3.8.1 AC Sources - Operating	A30 ---------- A39	A30 amended the Unit 1 TS to extend (on a ONE TIME basis) the Action Completion Time for diesel generator 1 B B from 72 hours to 10 days in order to allow the replacement of the diesel's electric generator. This change will NOT be incorporated into the Unit 2 TS. Since the changes per A30 were SUPERSEDED by A39, a markup is not provided for the change per A30. NRC approved A30 via letter dated 12/08/2000. ---------- A39 amended the Unit 1 TS to revise LCO 3.8.1's allowed outage time to restore an INOPERABLE emergency diesel generator to operable status from 72 hours to 14 days. This change will NOT be incorporated into the Unit 2 TS. NRC approved A39 via letter dated 07/01/2002.
TS 4.2 Reactor Core	A08 ------ A40 ------ A48 ------ A67	A08 amended the Unit 1 TS to provide for insertion of Lead Test Assemblies containing Tritium Producing Burnable Absorber Rods (TPBARs) during Cycle 2. TPBARS will NOT be used on Unit 2. NRC approved A08 via letter dated 09/15/1997. ---------- A40 amended the Unit 1 TS to allow Watts Bar to irradiate up to 2304 Tritium Producing Burnable Absorber Rods (TPBARs) in the reactor core each fuel cycle. TPBARS will NOT be used on Unit 2. NRC approved A40 via letter dated 09/23/2002. ---------- A48 amended the Unit 1 TS to revise the boron concentration requirements and limit the number of Tritium Producing Burnable Absorber Rods (TPBARs) that could be loaded and irradiated in the core to a corresponding value. TPBARS will NOT be used on Unit 2; thus, changes per A48 are not applicable to TS 4.2. NRC approved A48 via letter dated 10/08/2003. ---------- A67 amended the Unit 1 TS to revise the maximum number of TPBARs that can be irradiated in the Unit 1 reactor core to 400. TPBARS will NOT be used on Unit 2. NRC approved A67 via letter dated 01/18/2008.

TS SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS 5.7 Procedures, Programs, and Manuals	A27 ------ A38 ------ A63 ------ A70	A27 amended the Unit 1 TS to approve use of an alternate repair criterion (F*) in the tubesheet region of the steam generator. This change will NOT be applied to Unit 2 at this time. The portion of interest here (i.e., 5.7.2.12) was revised in entirety by A65; thus, no markup is included. NRC approved A27 via letter dated 09/08/2000. ---------- A38 amended the Unit 1 TS to incorporate voltage-based Alternate Repair Criteria for steam generator tubes. This change was NOT incorporated into the Unit 2 TS. The portion of interest here (i.e., 5.7.2.12) was revised in entirety by A65; thus, no markup is included. NRC approved A38 via letter dated 02/26/2002. ---------- A63 amended Unit 1 TS 5.7.2.19 to allow a ONE TIME, 5-year extension to the current 10-year test interval for the containment integrated leakage rate test. This change is NOT applicable to Unit 2; it was NOT incorporated in the Unit 2 TS. NRC approved A63 via letter dated 08/22/2006. ---------- A70 amended the Unit 1 TS to adopt TSTF Change Traveler TSTF-448, R3: Control Room Envelope Habitability. The changes per A70 were incorporated herein. NRC approved A70 via letter dated 10/08/2008.
TS 5.9 Reporting Requirements	A27 ------ A31 ------ A38	A27 amended the Unit 1 TS to approve use of an alternate repair criterion (F*) in the tubesheet region of the steam generator. This change will NOT be incorporated into the Unit 2 TS at this time. The portion of interest here (i.e., 5.9.9) was revised in entirety by A65; thus, no markup is included. NRC approved A27 via letter dated 09/08/2000. ---------- A31 amended the Unit 1 TS to approve Power Uprate using Leading Edge Flow Meter (LEFM). A31 changes for the portion of interest here (i.e., TS 5.9.5) will NOT be incorporated into the Unit 2 TS at this time. NRC approved A31 via letter dated 01/19/2001. ---------- A38 amended the Unit 1 TS to incorporate voltage-based Alternate Repair Criteria for steam generator tubes. This change was NOT incorporated into the Unit 2 TS. The portion of interest here (i.e., TS 5.9.9) was revised in entirety by A65; thus, no markup is included. NRC approved A38 via letter dated 02/26/2002.

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT	The PTLR is the unit specific document that provides the RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.9.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3459~~ 3411 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

(continued)

Not on U2

Use U2 Figure (for 3411 MWt)

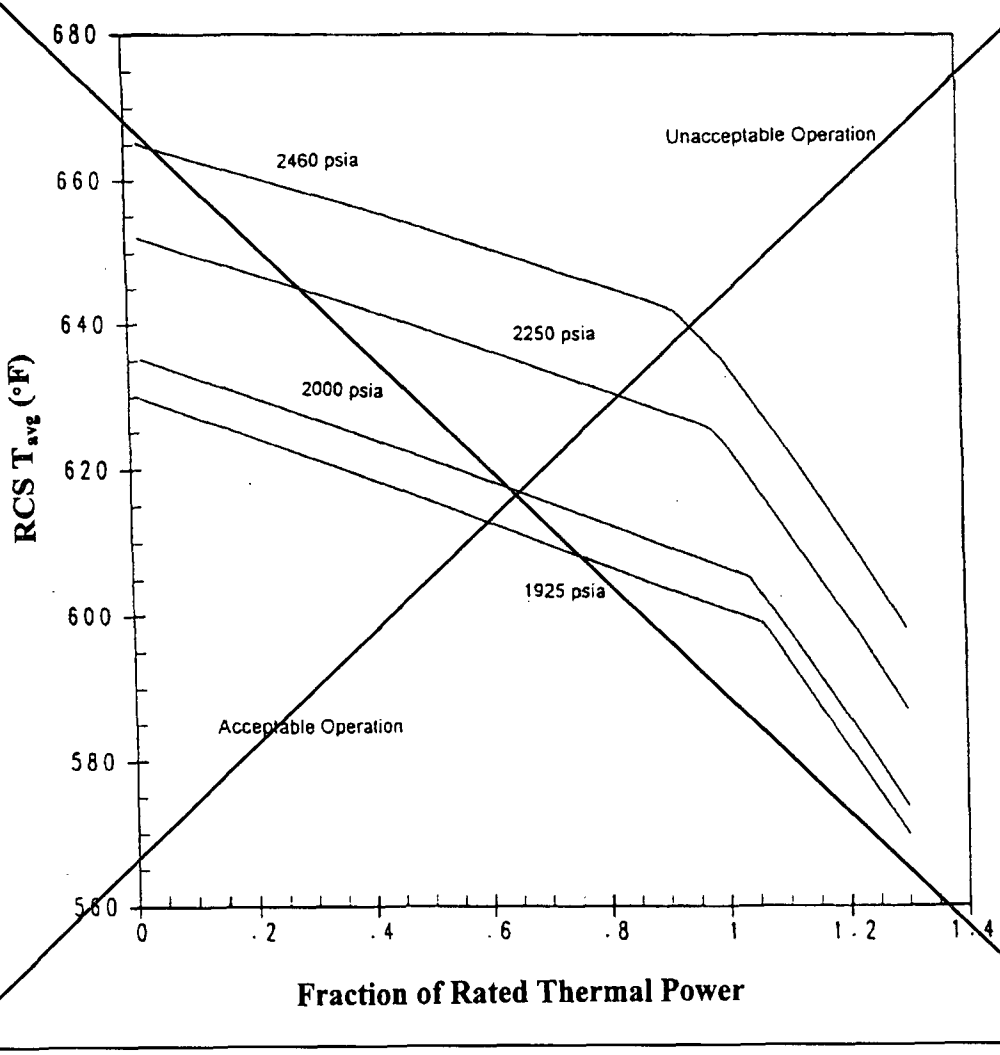

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

Not on U2

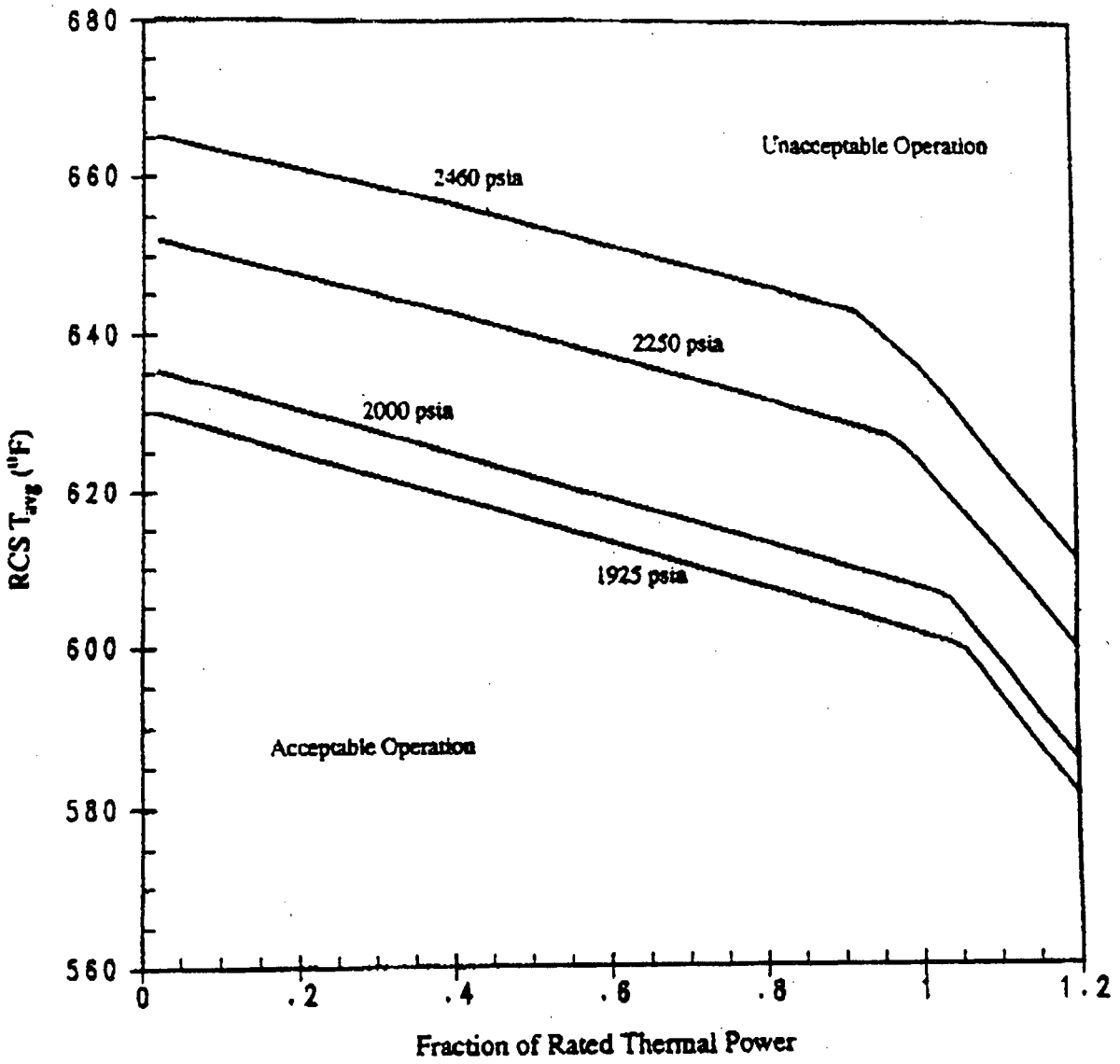

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

TSTF 418, R2

REPLACE WITH

12

ACTIONS (continued)

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

TSTF 418, R2

REPLACE WITH

72

78

72

TSTF 418, R2

REPLACE WITH

78

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	----------NOTE---------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------	
	E.1 Place channel in trip.	6 hours
	OR	
	E.2 Be in MODE 3.	12 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours
	OR	
	F.2 Increase THERMAL POWER to > P-10.	2 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
	AND	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. THERMAL POWER < P-6, one or two Intermediate Range Neutron Flux channels inoperable.	H.1 Restore channel(s) to OPERABLE status.	Prior to increasing THERMAL POWER to > P-6

TSTF 418, R2
REPLACE WITH
12

TSTF 418, R2
REPLACE WITH
72
78

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M. One channel inoperable.	----------NOTE---------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------	
	M.1 Place channel in trip.	6 hours
	OR	
	M.2 Reduce THERMAL POWER to < P-7.	12 hours
N. One Reactor ~~Coolant Flow--Low (single loop)~~ channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------	
	N.1 Place channel in trip.	6 hours
	OR	
	N.2 Reduce THERMAL POWER to < P-8.	10 hours

TSTF 418, R2
REPLACE WITH
12

TSTF 418, R2
REPLACE WITH
72
78

TSTF 169, R1
DELETE

TSTF 418, R2
REPLACE WITH
72
78

TSTF 169, R1
REPLACE WITH
P-7.

(continued)

A68

ACTIONS (continued)

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

TSTF-418, R2

REPLACE WITH

12

TSTF-418, R2

REPLACE WITH

72

76

TSTF-411, R1

REPLACE WITH

24

30

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Q. One RTB train inoperable.	~~NOTES~~ ~~1. One train may be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE.~~ ~~2. One RTB may be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE.~~	
	Q.1 Restore train to OPERABLE status.	1 hour
	OR	
	Q.2 Be in MODE 3.	7 hours
R. One channel inoperable.	R.1 Verify interlock is in required state for existing unit conditions.	1 hour
	OR	
	R.2 Be in MODE 3.	7 hours
S. One channel inoperable.	S.1 Verify interlock is in required state for existing unit conditions.	1 hour
	OR	
	S.2 Be in MODE 2.	7 hours

TSTF-411, R1
DELETE

TSTF-411, R1
REPLACE WITH
NOTE

TSTF-411, R1
REPLACE WITH
4

TSTF-411, R1
REPLACE WITH
24 hours

30

(continued)

468

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
T. One trip mechanism inoperable for one RTB.	T.1 Restore inoperable trip mechanism to OPERABLE status.	48 hours
	OR	
	T.2.1 Be in MODE 3.	54 hours
	AND	
	T.2.2 Open RTB.	55 hours
U. One Steam Generator Water Level Low-Low channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------	
	U.1.1 Place channel in trip.	6 hours
	AND	
	U.1.2 For the affected protection set, set the Trip Time Delay (T_S) to match the Trip Time Delay (T_M).	6 hours
	OR	
	U.2 Be in MODE 3.	12 hours

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
72
72
78

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
V. One Vessel ΔT channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------	
	V.1 Set the Trip Time Delay threshold power level for (T_S) and (T_M) to 0% power.	6 hours
	OR	
	V.2 Be in MODE 3.	12 hours
W. One channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------	
	W.1 Place channel in trip.	6 hours
	OR	
	W.2 Be in MODE 3.	12 hours
X. One channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------	
	X.1 Place channel in trip.	6 hours
	OR	
	X.2 Reduce THERMAL POWER to < P-7.	12 hours

TSTF-418, R2

REPLACE WITH

72

78

TSTF-418, R2

REPLACE WITH

12

TSTF-418, R2

REPLACE WITH

72

78

TSTF-418, R2

REPLACE WITH

72

78

(continued)

A68

ACTIONS (continued)

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

TSTF-418, R2

REPLACE WITH
72
76

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	NOTES 1. Adjust NIS channel if absolute difference is > 2%. 2. Required to be performed within 12 hours after THERMAL POWER is ≥ 15% RTP. Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours

(continued)

A68

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	----------NOTES---------- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Required to be performed within 96 hours after THERMAL POWER is ≥ 15% RTP. ---------- Compare results of the incore detector measurements to NIS AFD.	31 effective full power days (EFPD)
SR 3.3.1.4	----------NOTE---------- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. ---------- Perform TADOT.	31 days on a STAGGERED TEST BASIS TSTF-411, R1 REPLACE WITH 62
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS TSTF-411, R1 REPLACE WITH 92
SR 3.3.1.6	----------NOTE---------- Required to be performed within 6 days after THERMAL POWER is ≥ 50% RTP. ---------- Calibrate excore channels to agree with incore detector measurements.	92 EFPD

(continued)

A68

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.7	----------NOTE---------- For Functions 2 and 3 (Power Range Instrumentation), this Surveillance shall include verification that interlock P-10 is in the required state for existing unit conditions. ---------- Perform COT.	92 days TSTF-411, R1 REPLACE WITH 184
SR 3.3.1.8	----------NOTES---------- 1. Not required to be performed for Source Range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 2. This Surveillance shall include verification that interlock P-6 is in the required state for existing unit conditions. ---------- Perform COT.	------NOTE----- Only required when not performed within previous 31 days ---------- Prior to reactor startup AND Four hours after reducing power below P-10 for intermediate range instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND Every 31 days thereafter

(continued)

A68

SURVEILLANCE REQUIREMENTS (continued)

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

TSTF-311, R0

REPLACE WITH

Prior to exceeding the P-9 interlock whenever the unit has been in MODE 3, if not performed within the previous 31 days

A68

TSTF-169, R1

REPLACE WITH

1 (f)

Table 3.3.1-1 (page 3 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
9. Pressurizer Water Level-High	1 (f)	3	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 92.7% span	92% span
10. Reactor Coolant Flow-Low						
a. Single Loop	1 (g)	3 per loop	N	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.7% flow	90% flow
b. Two Loops	1 (h)	3 per loop	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.7% flow	90% flow
11. Undervoltage RCPs	1 (f)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4734 V	4830 V
12. Underfrequency RCPs	1 (f)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.9 Hz	57.5 Hz

TSTF-169, R1 - MOVE INFORMATION OF BOX UP TO HERE

TSTF-169, R1

DELETE

(continued)

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

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	C.1 --------NOTE--------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------- Restore train to OPERABLE status.	6 hours
	OR	
	C.2.1 Be in MODE 3.	12 hours
	AND	
	C.2.2 Be in MODE 5.	42 hours
D. One channel inoperable.	D.1 --------NOTE--------- One channel may be bypassed for up to 4 hours for surveillance testing. ---------- Place channel in trip.	6 hours
	OR	
	D.2.1 Be in MODE 3.	12 hours
	AND	
	D.2.2 Be in MODE 4.	18 hours

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
24
30
60

TSTF-418, R2
REPLACE WITH
72
78
84

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One Containment Pressure channel inoperable. TSTF-418, R2 DELETE TSTF-418, R2 REPLACE WITH 12	E.1 --------NOTE-------- One ~~additional~~ channel may be bypassed for up to 4 hours for surveillance testing. ---------- Place channel in bypass.	6 hours
	OR	
	E.2.1 Be in MODE 3.	12 hours
	AND	
	E.2.2 Be in MODE 4.	18 hours
F. One channel or train inoperable.	F.1 Restore channel or train to OPERABLE status.	48 hours
	OR	
	F.2.1 Be in MODE 3.	54 hours
	AND	
	F.2.2 Be in MODE 4.	60 hours

TSTF-418, R2
REPLACE WITH
72
78
84

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. One train inoperable.	G.1 --------NOTE-------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------- Restore train to OPERABLE status.	6 hours
	OR	
	G.2.1 Be in MODE 3.	12 hours
	AND	
	G.2.2 Be in MODE 4.	18 hours
H. One train inoperable.	H.1 --------NOTE-------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---------- Restore train to OPERABLE status.	6 hours
	OR	
	H.2.1 Be in MODE 3.	12 hours
	AND	
	H.2.2 Be in MODE 4.	18 hours

TSTF-418, R2	
REPLACE WITH	
24	
30	
36	
24	
30	
36	

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One Steam Generator Water Level--High High channel inoperable.	I.1 --------NOTE-------- One channel may be bypassed for up to 4 hours for surveillance testing. ---------- Place channel in trip.	6 hours
	OR	
	I.2.1 Be in MODE 3.	12 hours
	OR	
	I.2.2 Be in MODE 4.	18 hours
J. One Main Feedwater Pumps trip channel inoperable.	J.1 Restore channel to OPERABLE status.	48 hours
	OR	
	J.2 Be in MODE 3.	54 hours
K. One channel inoperable.	K.1 --------NOTE-------- One ~~additional~~ channel may be bypassed for up to 4 hours for surveillance testing. ---------- Place channel in bypass.	6 hours
	OR	

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
72
78
84

TSTF-418, R2
DELETE

TSTF-418, R2
REPLACE WITH
72

(continued)

A68

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. (continued)	K.2.1 Be in MODE 3.	12 hours
	AND	
	K.2.2 Be in MODE 5.	42 hours
L. One P-11 interlock channel inoperable.	L.1 Verify interlock is in required state for existing unit condition.	1 hour
	OR	
	L.2.1 Be in MODE 3.	7 hours
	AND	
	L.2.2 Be in MODE 4.	13 hours

TSTF-418, R2

REPLACE WITH
78
108

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	TSTF-418, R2 REPLACE WITH
M. One Steam Generator Water Level--Low--Low channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------		
	M.1.1 Place channel in trip.	6 hours	72
	AND		
	M.1.2 For the affected protection set, set the Trip Time Delay (T_e) to match the Trip Time Delay (T_m)	6 hours	72
	OR		
	M.2.1 Be in MODE 3.	12 hours	78
	AND		
	M.2.2 Be in MODE 4.	18 hours	84
N. One Vessel ΔT channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 4 hours for surveillance testing. ----------		
	N.1 Set the Trip Time Delay threshold power level for (T_e) and (T_m) to 0% power.	6 hours	72
	OR		
	N.2 Be in MODE 3.	12 hours	78

TSTF-418, R2 REPLACE WITH 12

(continued)

A68

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
O. One MSVV Room Water Level High channel inoperable	----------NOTE---------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----------	
	O.1 Place channel in trip	6 hours
	OR	
	O.2 Be in MODE 3	12 hours

TSTF-418, R2

REPLACE WITH
12

TSTF-418, R2

REPLACE WITH
72
78

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

SURVEILLANCE		FREQU
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	31 days o STAGGERED BASIS
SR 3.3.2.3	Perform MASTER RELAY TEST.	31 days o STAGGERED BASIS
SR 3.3.2.4	Perform COT.	92 days

TSTF-411, R1

REPLACE WITH
92
92
184

(continued)

A68

Table 3.3.2-1 (page 3 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
c. Containment Pressure-High High	1, 2[c], 3[c]	4	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 2.9 psig	2.8 psig
d. Steam Line Pressure						
(1) Low	1, 2[c], 3[a] [c]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR .3.2.10	≥ 666.6[b] psig	675[b] psig
(2) Negative Rate-High	3[d] [c]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 108.5[e] psi	100[e] psi
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1, 2[f], 3[f]	2 trains	H	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-High High(P-14)	1, 2[f], 3[f]	3 per SG	I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10 [h]	≤ 83.1%	82.4%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. North MSV Vault Room Water Level - High	1, 2[f], [g]	3/vault Room	O	SR 3.3.2.6 SR 3.3.2.9	≤ 5.31 inches	4 inches
e. South MSV Vault Room Water Level - High	1, 2[f], [g]	3/vault Room	O	SR 3.3.2.6 SR 3.3.2.9	≤ 4.56 inches	4 inches

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.
(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(c) Except when all MSIVs are closed and de-activated.
(d) Function automatically blocked above P-11 (Pressurizer Interlock) setpoint and is enabled below P-11 when safety injection on Steam Line Pressure Low is manually blocked.
(e) Time constants utilized in the rate/lag controller are t_3 and $t_4 \geq 50$ seconds.
(f) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.
(g) MODE 2 if Turbine Driven Main Feed Pumps are operating.
~~(h) For the time period between February 23, 2000, and prior to turbine restart (following the next time the turbine is removed from service), the response time test requirement of SR 3.3.2.10 is not applicable for 1-FSV-47-027.~~

Not on U2

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

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
1. Reactivity Control	
a. Source Range Neutron Flux	1
b. Reactor Trip Breaker Position Indication	1 per trip breaker
2. Reactor Coolant System (RCS) Pressure Control	
a. Pressurizer Pressure Indication or RCS Wide Range Pressure Indication	1
b. Pressurizer Power Operated Relief Valve (PORV) Control and Pressurizer Block Valve Control	1 each per relief path
c. Pressurizer Heater Control	1
3. RCS Inventory Control	
a. Pressurizer Level Indication	1
b. Charging and Letdown Flow Control and Indication	1
4. Decay Heat Removal via Steam Generators (SGs)	
a. RCS Hot Leg Temperature Indication	1 per loop (Refer to Note A below)
b. AFW Controls	1
c. SG Pressure Indication and Control	1 per SG
d. SG Level Indication and AFW Flow Indication	1 per SG
e. SG T_{sat} Indication	1 per SG
5. Decay Heat Removal via RHR System	
a. RHR Flow Control	1
b. RHR Temperature Indication	1

~~Notes:~~
~~A. For Function 4a, the temperature indicator for RCS hot leg 4 is not required to be operable for the remainder of Cycle 6.~~

Not on U2

TSTF-411, R1

INSERT

NOTE

This surveillance is only applicable to the actuation logic of the ESFAS instrumentation.

Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Vent Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	Perform ACTUATION LOGIC TEST.	31 days on STAGGERED T BASIS
SR 3.3.6.3	Perform MASTER RELAY TEST.	31 days on STAGGERED T BASIS
SR 3.3.6.4	Perform COT.	92 days
SR 3.3.6.5	Perform SLAVE RELAY TEST.	92 days OR 18 months for Westinghouse type AR relays
SR 3.3.6.6	NOTE Verification of setpoint is not required. Perform TADOT.	18 months
SR 3.3.6.7	Perform CHANNEL CALIBRATION.	18 months

TSTF-411, R1

REPLACE WITH

92

92

A68

TSTF-411, R1

INSERT

NOTE

This surveillance is only applicable to the master relays of the ESFAS instrumentation.

Was "≥ 6%"

SURVEILLANCE		FREQUENCY
SR 3.4.5.2	Verify steam generator secondary side water levels are ~~greater than or equal to 32%~~ narrow range for required RCS loops.	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

Not on U2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.6.2	Verify one 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

Was "6%"

Not on U2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least two steam generators (SGs) shall be greater than or equal to ~~32%~~ narrow range.

Was "6%"

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

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

2. No reactor coolant pump shall be started with one or more RCS cold leg temperatures less than or equal to 350°F unless the secondary side water temperature of each SG is less than or equal to 50°F above each of the RCS cold leg temperatures.

3. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. AND Required SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately Immediately

(continued)

Not on UL

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration. AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	12 hours
SR 3.4.7.2	Verify SG secondary side water level is greater than or equal to ~~32%~~ narrow range in required SGs.	12 hours
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

Was "6%"

Not on U2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 7630 gallons and ≤ 8000 gallons.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 610 psig and ≤ 660 psig	12 hours
SR 3.5.1.4	~~NOTE~~ ~~The number of TPBARs in the reactor core is contained in the Core Operating Limits Report (COLR) for each operating cycle.~~ Verify boron concentration in each accumulator is ~~as provided below depending on the number of tritium-producing burnable absorber rods (TPBARs) installed in the reactor core for this operating cycle:~~ ~~Number of TPBARs~~ — ~~Boron Concentration Ranges~~ ~~0-400~~ — ≥ 3000 ppm and ≤ 3300 ppm.	31 days <u>AND</u> -------NOTE ---------- Only required to be performed for affected accumulators. ---------- Once within 6 hours after each solution volume increase of ≥ 75 gallons, that is not the result of addition from the refueling water storage tank.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. NumberPositionFunction FCV-63-1 Open RHR Supply FCV-63-22 Open SIS Discharge	12 hours
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.3	Verify ECCS piping is full of water.	31 days ~~NOTE: Surveillance performance not required for safety injection hot leg injection lines until start up from the Fall 2003 refueling outage.~~
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months

(continued)

Not on U2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	----------NOTE---------- Only required to be performed when ambient air temperature is < 60°F or > 105°F. ---------- Verify RWST borated water temperature is ≥ 60°F and ≤ 105°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 370,000 gallons.	7 days
SR 3.5.4.3	~~NOTE~~ ~~The number of TPBARs in the reactor core is contained in the Core Operating Limits Report (COLR) for each operating cycle.~~ Verify boron concentration in the RWST is ~~as provided below depending on the number of tritium producing burnable absorber rods (TPBARs) installed in the reactor core for this operating cycle:~~ ~~Number of TPBARS~~ | ~~Boron Concentration Ranges~~ ~~0-400~~ | ≥ 3100 ppm and ≤ 3300 ppm	7 days

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.1 and ≤ +0.3 psid relative to the annulus.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~NOTE~~ ~~When opening or closing Penetration 1-EQH-271-0010 or 1-EQH-271-0011 in the Shield Building Dome during Cycle 7 operation, time is allowed for Containment Annulus pressure equalization to occur.~~ Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

Not on U2

3.6 CONTAINMENT SYSTEMS

3.6.8 Hydrogen Mitigation System (HMS)

LCO 3.6.8 Two HMS trains shall be OPERABLE. ~~(* See Note below)~~

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One HMS train inoperable. ~~* See Note below~~	A.1 Restore HMS train to OPERABLE status.	7 days
	OR	
	A.2 Perform SR 3.6.8.1 on the OPERABLE train.	Once per 7 days
B. One containment region with no OPERABLE hydrogen ignitor. ~~* See Note below~~	B.1 Restore one hydrogen ignitor in the affected containment region to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

~~* NOTE~~

~~For the time period between June 9, 1998, and the next WBN Unit 1 entry into MODE 3, HMS Train A is considered OPERABLE with 32 of 34 ignitors OPERABLE. The following additional CONDITION and REQUIRED ACTION applies:~~

~~CONDITION~~

~~Reactor Cavity Region (Hydrogen Ignitors 30A and 46B) and Steam Generator No. 4 Enclosure Lower Compartment Region (Hydrogen Ignitors 31A and 45B) with no OPERABLE hydrogen ignitor.~~

~~REQUIRED ACTION/COMPLETION TIME~~

~~Restore one hydrogen ignitor in each region to OPERABLE status within 72 hours.~~

Not on U2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR	3.6.8.1	Energize each HMS train power supply breaker and verify ≥33~~*~~ ignitors are energized in each train. ~~* See Note below~~	92 days~~*~~
SR	3.6.8.2	Verify at least one hydrogen ignitor is OPERABLE in each containment region.	92 days
SR	3.6.8.3	Energize each hydrogen ignitor and verify temperature is ≥1700°F.	18 months

~~* NOTE~~

~~For the time period between June 9, 1998, and the next WBN Unit 1 entry into MODE 3, SR 3.6.8.1 shall verify ≥ 32 ignitors are OPERABLE on HMS Train A at a frequency of 46 days.~~

Not on Unit 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.11.2	Verify total weight of stored ice is greater than or equal to ~~2,404,500~~ lb by: Was "2,158,000" a. Weighing a representative sample of ≥ 144 ice baskets and verifying each basket contains greater than or equal to ~~1237~~ lb of ice; and Was "1110" b. Calculating total weight of stored ice, at a 95 percent confidence level, using all ice basket weights determined in SR 3.6.11.2.a.	18 months
SR 3.6.11.3	Verify azimuthal distribution of ice at a 95 percent confidence level by subdividing weights, as determined by SR 3.6.11.2.a, into the following groups: a. Group 1-bays 1 through 8; b. Group 2-bays 9 through 16; and c. Group 3-bays 17 through 24. The average ice weight of the sample baskets in each group from radial rows 1, 2, 4, 6, 8, and 9 shall be greater than or equal to ~~1237~~ lb.	18 months
SR 3.6.11.4	Verify, by visual inspection, accumulation of ice on structural members comprising flow channels through the ice bed is less than or equal to 15 percent blockage of the total flow area for each safety analysis section.	18 months

(continued)

Not on U2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.12.3	Verify, by visual inspection, each inlet door is not impaired by ice, frost, or debris.	~~-----NOTE------ The 3 month performance due September 9, 1996 (per SR 3.0.2) may be extended until October 21, 1996. ----------~~ 3 months during first year after receipt of license AND 18 months
SR 3.6.12.4	Verify torque required to cause each inlet door to begin to open is ≤ 675 in-lb.	~~-----NOTE------ The 3 month performance due September 9, 1996 (per SR 3.0.2) may be extended until October 21, 1996. ----------~~ 3 months during first year after receipt of license AND 18 months

(continued)

Not on U2

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.6.12.5	Perform a torque test on a sampling of ≥ 50% of the inlet doors.	~~-----NOTE------ The 3 month performance due September 9, 1996 (per SR 3.0.2) may be extended until October 21, 1996. ----------~~ 3 months during first year after receipt of license AND 18 months
SR 3.6.12.6	Verify for each intermediate deck door: a. No visual evidence of structural deterioration; b. Free movement of the vent assemblies; and c. Free movement of the door.	3 months during first year after receipt of license AND 18 months

(continued)

Not on U2

3.6 CONTAINMENT SYSTEMS

3.6.15 Shield Building

LCO 3.6.15 The Shield Building shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Shield Building inoperable.	A.1 Restore Shield Building to OPERABLE status.	24 hours
B. ------NOTE--------- Annulus pressure requirement is not applicable during ventilating operations, required annulus entries, or Auxiliary Building isolations ~~not exceeding~~ 1 hour in duration ~~or while Penetration 1-EQH-271-0010 or 1-EQH-271-0011 in the Shield Building dome is open until annulus pressure is restored.*~~ ---------- Annulus pressure not within limits.	B.1 Restore annulus pressure within limits.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

~~*1. The combined opening time of Penetrations 1-EQH-271-0010 or 1-EQH-271-0011 is limited to a total time of five hours a day, six days a week during Cycle 7 operation.~~

2. Penetrations 1-EQH-271-0010 ~~or 1-EQH-271-0011 in the Shield~~ Building Dome may not be opened if in Action Conditions LCO 3.6.9A or 3.8.1B.

3. Upon opening Penetration 1-EQH-271-0010 or 1-EQH-271-0011 in the ~~Shield~~ Building Dome, both EGTS control loops shall be placed in the A-Auto Stand-by position and returned to ~~normal position following closure of penetration.~~

Not on U2

3.7 Plant Systems

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Five MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam generators with one MSSV inoperable.	A.1 Reduce THERMAL POWER to ≤ ~~58~~ % RTP. Was "59"	4 hours
B. One or more steam generators with two or more MSSVs inoperable.	B.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. AND --------NOTE---------- Only required in MODE 1 --- B.2 Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours 36 hours
C. Required Action and associated Completion Time not met. OR One or more steam generators with ≥ 4 MSSVs inoperable.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

Not on U2

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
3	≤ ~~41~~ Was "42"
2	≤ ~~25~~ Was "26"

Not U2

3.7 PLANT SYSTEMS

3.7.8 Essential Raw Cooling Water (ERCW) System

LCO 3.7.8 Two ERCW trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ERCW train inoperable~~, other than for Condition C.~~	A.1 ----------NOTES---------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources- Operating," for emergency diesel generator made inoperable by ERCW. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops-MODE 4," for residual heat removal loops made inoperable by ERCW. ---------- Restore ERCW train to OPERABLE status.	72 hours

(continued)

Not for U2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
~~C. Two Train A ERCW pumps (A-A and B-A) inoperable and two Train A ERCW pumps operable (C-A and D-A). *~~	~~C.1 Align the operable pumps (C-A and D-A) to concurrently autostart from the 2A-A 6.9 KV Shutdown Board.~~ ~~AND~~ ~~C.2 Restore at least one of the pumps (A-A or B-A) to OPERABLE status.~~	~~72 hours~~ ~~10 days~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	----------NOTE---------- Isolation of ERCW flow to individual components does not render the ERCW inoperable. ---------- Verify each ERCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

~~* This CONDITION will apply until the A-A or B-A pump is repaired and declared operable or until July 31, 2008, whichever occurs first.~~

Not for UL

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

Insert 1

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6
During movement of irradiated fuel assemblies.

ACTIONS

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. One CREVS train inoperable.~~	~~A.1 Restore CREVS train to OPERABLE status.~~	~~7 days~~
~~B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~
~~C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.~~	~~C.1 Place OPERABLE CREVS train in emergency mode.~~ ~~OR~~ ~~C.2 Suspend movement of irradiated fuel assemblies.~~	~~Immediately~~ ~~Immediately~~

~~(continued)~~

Insert 2

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ACTIONS (continued)

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~D. Two CREVS trains inoperable in MODE 1, 2, 3, or 4 due to actions taken as a result of a tornado warning.~~	~~D.1 Restore one CREVS train to OPERABLE status.~~	~~8 hours~~
~~E. Two CREVS trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.~~	~~E.1 Suspend movement of irradiated fuel assemblies.~~	~~Immediately~~
~~F. Two CREVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition D.~~	~~F.1 Enter LCO 3.0.3.~~	~~Immediately~~

Insert 3

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.7.10.1~~	~~Operate each CREVS train for ≥ 15 minutes.~~	~~31 days~~
~~SR 3.7.10.2~~	~~Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).~~	~~In accordance with VFTP~~

(continued)

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~~SURVEILLANCE REQUIREMENTS (continued)~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.7.10.3~~	~~Verify each CREVS train actuates on an actual or simulated actuation signal.~~	~~18 months~~
~~SR 3.7.10.4~~	~~Verify one CREVS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the outside atmosphere and adjacent areas during the pressurization mode of operation at a makeup flow rate of ≤ 711 cfm and a recirculation flow rate ≥ 2960 and ≤ 3618 cfm.~~	~~18 months on a STAGGERED TEST BASIS~~

Insert 5

Insert 4

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WBN Technical Specification (TS) Change TS-07-14
Inserts for Proposed Technical Specification Changes

The "inserts" below are annotated to reflect the changes and additions that are based on TSTF-448. The deletions are shown as strikethrough text and the additions are shown as bold-italicized text.

Insert 1:

----------***NOTE***----------
The control room envelope (CRE) boundary may be opened intermittently under administrative control.

Insert 2:

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One CREVS train inoperable ***for reasons other than Condition B.***	A.1	Restore CREVS train to OPERABLE status.	7 days
B.	***One or more CREVS trains inoperable due to inoperable CRE boundary in Mode 1, 2, 3, or 4.***	***B.1***	***Initiate action to implement mitigating actions.***	***Immediately***
		AND		
		B.2	***Verify mitigating actions ensure CRE occupant exposures to radiological and chemical hazards will not exceed limits and CRE occupants are protected from smoke hazards."***	***24 hours***
		AND		
		B.3	***Restore CRE boundary to OPERABLE status.***	***90 days***

(continued)

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Insert 3:

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C~~B. Required Action and associated Completion Time of Condition A ***or B*** not met in MODE 1, 2, 3, or 4.	~~C~~B.1 Be in MODE 3. AND ~~C~~B.2 Be in MODE 5.	6 hours 36 hours
~~D~~C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	~~D~~C.1 Place OPERABLE CREVS train in emergency mode. OR ~~D~~C.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
~~E~~D. Two CREVS trains inoperable in MODE 1, 2, 3, or 4 due to actions taken as a result of a tornado warning.	~~E~~D.1 Restore one CREVS train to OPERABLE status.	8 hours
~~F~~E. Two CREVS trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies. ***OR*** ***One or more CREVS trains inoperable due to inoperable CRE boundary in Mode 5 or 6, or during movement of irradiated fuel assemblies.***	~~F~~E.1 Suspend movement of irradiated fuel assemblies.	Immediately

(continued)

A70

WBN Technical Specification (TS) Change TS-07-14
Inserts for Proposed Technical Specification Changes

Insert 4:

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~G.~~ Two CREVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition ***B or E~~D~~***.	~~G.~~1 Enter LCO 3.0.3.	Immediately

Insert 5:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CREVS train for ≥ 15 minutes.	31 days
SR 3.7.10.2	Perform required CREVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with ***the*** VFTP
SR 3.7.10.3	Verify each CREVS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	~~Verify one CREVS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the outside atmosphere and adjacent areas during the pressurization mode of operation at a makeup flow rate of ≤ 711 cfm and a recirculation flow rate ≥ 2960 and ≤ 3618 cfm.~~ ***Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Habitability Program.***	~~18 months on a STAGGERED TEST BASIS~~ ***In accordance with the Control Room Envelope Habitability Program***

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Restore offsite circuit to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO
B. One required DG inoperable.	B.1 Perform SR 3.8.1.1 for the offsite circuits.	1 hour AND Once per 8 hours thereafter
	AND	
	B.2 Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND	
	B.3.1 Determine OPERABLE DGs are not inoperable due to common cause failure. OR	12 hours
	B.3.2 Perform SR 3.8.1.2 for OPERABLE DGs.	12 hours
	AND	
		(continued)

Not on U2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. (continued)~~	~~B.4 Restore required DG to OPERABLE status.~~	~~14 days~~ ~~AND~~ ~~17 days from discovery of failure to meet LCO~~
B ~~C.~~ ~~Two~~ required DGs in Train A inoperable. OR ~~Two~~ required DGs in Train B inoperable. Was "One or more"	~~C~~ B.1 Perform SR 3.8.1.1 for the offsite circuits. AND ~~C~~ B.2 Declare required feature(s) supported by the inoperable DGs inoperable when its required redundant feature(s) is inoperable. AND ~~C~~ B.3.1 Determine OPERABLE DGs are not inoperable due to common cause failure. OR B ~~C~~.3.2 Perform SR 3.8.1.2 for OPERABLE DGs. AND	1 hour AND Once per 8 hours thereafter 4 hours from discovery of Condition ~~C~~ B concurrent with inoperability of redundant required feature(s) ~~12~~ 24 hours ~~12~~ hours (continued)

Not on U2

CONDITION	REQUIRED ACTION	COMPLETION TIME
B ~~C~~. (continued)	~~C~~ B .4 Restore at least one required DG to OPERABLE status.	72 hours AND 6 days from discovery of failure to meet LCO
C ~~D~~. Two offsite circuits inoperable.	~~D~~ C .1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND ~~D~~ C .2 Restore one offsite circuit to OPERABLE status.	12 hours from discovery of Condition ~~D~~ C concurrent with inoperability of redundant required features 24 hours

(continued)

Not on U2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E.~~ D One offsite circuit inoperable. AND One or more required DG(s) in Train A inoperable. OR One or more required DG(s) in Train B inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating," when Condition ~~E~~ D is entered with no AC power source to any train. ---------- ~~E~~ D.1 Restore offsite circuit to OPERABLE status. OR ~~E~~ D.2 Restore required DG(s) to OPERABLE status.	 12 hours 12 hours
~~F.~~ E One or more required DG(s) in Train A inoperable. AND One or more required DG(s) in Train B inoperable.	~~F~~ E.1 Restore required DGs in Train A to OPERABLE status. OR ~~F~~ E.2 Restore required DGs in Train B to OPERABLE status	2 hours 2 hours
F ~~G.~~ Required Action and Associated Completion Time of Condition A, B, C, D, E, or F not met.	~~G~~ F.1 Be in MODE 3. AND ~~G~~ F.2 Be in MODE 5.	6 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~H~~ G. Two offsite circuits inoperable. AND One or more required DG(s) in Train A inoperable. OR One or more required DG(s) in Train B inoperable.	~~H~~ G.1 Enter LCO 3.0.3.	Immediately
H ~~I~~. One offsite circuit inoperable. AND One or more required DG(s) in Train A inoperable. AND One or more required DG(s) in Train B inoperable.	~~I~~ H.1 Enter LCO 3.0.3.	Immediately

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries

The site and exclusion area boundaries shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2 (within the 3-mile circle).

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or Zirlo fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. ~~For Unit 1, Watts Bar is authorized to place a maximum of 704 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.~~

4.2.2 Control Rod Assemblies

The reactor core shall contain 57 control rod assemblies. The control material shall be boron carbide with silver indium cadmium tips as approved by the NRC.

(continued)

Not on UL

5.7 Procedures, Programs, and Manuals

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

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

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

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

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.7.2.19 Containment Leakage Rate Testing Program

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

~~The Fall 2007 end date for conducting the 10 year interval containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than Fall 2012.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 15.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

(continued)

Not on UL

5.7 Procedures, Programs, and Manuals

5.7.2.19 Containment Leakage Rate Testing Program (continued)

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the combined Type B and Type C tests, and ≤ 0.75 L_a for Type A tests.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

2) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 6 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

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

[5.7.2.20]

Insert 6

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Insert 6:

5.2.7.20 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate defined in the Ventilation Filter Testing Program (VFTP), at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

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5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

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

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

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. ~~When an initial assumed power level of 102 percent of rated thermal power is specified in a previously approved method, 100.6 percent of rated thermal power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 6 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102 percent of rated thermal power (3411 MWt) shall be used.~~

~~The approved analytical methods are~~ specifically those described in the following documents:

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

2a. WCAP-12945-P-A, Volume I (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985. Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

(continued)

 Not on U2

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

5. WCAP-15088-P, Rev. 1, "Safety Evaluation Supporting A More Negative EOL Moderator Temperature Coefficient Technical Specification for the Watts Bar Nuclear Plant," July 1999, (W Proprietary), as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 20 (Methodology for Specification 3.1.4 - Moderator Temperature Coefficient.).

~~6. Caldon, Inc. Engineering Report 80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997; and Caldon, Inc. Engineering Report 160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™," Revision 0, May 2000; as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 31.~~

7. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

8. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

9. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999. (Methodology for Specification 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

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

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

(continued)

Not on UL

ENCLOSURE 3

Unit 2 TS Bases Review Matrix: Unit 1 Revisions 86 through 91 With Associated Markups

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

UNIT 2 TS BASES REVIEW MATRIX: UNIT 1 REVISIONS 86 THROUGH 91

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.3.1 Reactor Trip System (RTS) Instrumentation	R60 (A47) ---------- R90 (A68)	R60 revised the Unit 1 TS Bases to implement the changes made by A47 to Unit 1 TS (allow an alternate method for the measurement of RCS total flow rate via measurement of the RCS elbow tap differential pressures). Developmental Revision A for the Unit 2 TS inappropriately stated, "The changes will be applied to Unit 2." NRC approved A47 via letter dated 10/03/2003. ********** DISCREPANCY: The changes per R60 were inappropriately incorporated into Developmental Revision A of the Unit 2 TS. Developmental Revision B of the Unit 2 TS Bases deleted the changes incorporated into the Unit 2 TS Bases per R60 (A47); thus, an additional markup is NOT being provided. ---------- R90 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A68 (Allow relaxations of various logic completion times, bypass test times, allowable outage times, and surveillance testing intervals previously reviewed and approved by NRC under WCAP-14333-P-A, "Probabilistic Risk Analysis of RPS and ESFAS Test Times and Completion Times," and WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times." R90 also incorporated TSTF-169, "Deletion of Condition 3.3.1.N," and TSTF-311, "Revision of Surveillance Frequency for TADOT on Turbine Trip Functional Unit."). The changes per R90 were incorporated herein. NRC approved A68 via letter dated 06/30/2008.
TS Bases 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation	R30 (A23) ---------- R90 (A68)	R30 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A23 (ONE TIME exception to allow for relief from response time testing 1-FSV-47-027). This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS Bases. NRC approved A23 via letter dated 03/22/2000. ---------- R90 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A68 (Allow relaxations of various logic completion times, bypass test times, allowable outage times, and surveillance testing intervals previously reviewed and approved by NRC under WCAP-14333-P-A, "Probabilistic Risk Analysis of RPS and ESFAS Test Times and Completion Times," and WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times." R90 also incorporated TSTF-169, "Deletion of Condition 3.3.1.N," and TSTF-311, "Revision of Surveillance Frequency for TADOT on Turbine Trip Functional Unit."). The changes per R90 were incorporated herein. NRC approved A68 via letter dated 06/30/2008.

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.3.6 Containment Vent Isolation Instrumentation	R87 ---------- R90 (A68)	R87 revised the Unit 1 TS Bases because of DCN 52220-A: tied the ABI and CVI signals together so that either signal initiates the other signal. The changes per R87 were incorporated herein. R87 was provided to the NRC via letter dated 09/22/2008. ---------- R90 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A68 (Allow relaxations of various logic completion times, bypass test times, allowable outage times, and surveillance testing intervals previously reviewed and approved by NRC under WCAP-14333-P-A, "Probabilistic Risk Analysis of RPS and ESFAS Test Times and Completion Times," and WCAP-15376-P-A, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times." R90 also incorporated TSTF-169, "Deletion of Condition 3.3.1.N," and TSTF-311, "Revision of Surveillance Frequency for TADOT on Turbine Trip Functional Unit."). The changes per R90 were incorporated herein. NRC approved A68 via letter dated 06/30/2008.
TS Bases 3.3.8 Auxiliary Building Gas Treatment System (ABGTS) Actuation Instrumentation	R87	R87 revised the Unit 1 TS Bases because of DCN 52220-A: tied the ABI and CVI signals together so that either signal initiates the other signal. The changes per R87 were incorporated herein. R87 was provided to the NRC via letter dated 09/22/2008.
TS Bases 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	R60 (A47)	R60 revised the Unit 1 TS Bases to implement the changes made by A47 to the Unit 1 TS (allow an alternate method for the measurement of RCS total flow rate via measurement of the RCS elbow tap differential pressures). Developmental Revision A for the Unit 2 TS inappropriately stated, "The changes will be applied to Unit 2." NRC approved A47 via letter dated 10/03/2003. ********** DISCREPANCY: The changes per R60 were inappropriately incorporated into Developmental Revision A of the Unit 2 TS. Developmental Revision B of the Unit 2 TS Bases deleted the changes incorporated per R60 (A47); thus, an additional markup is NOT being provided.
TS Bases 3.4.5 RCS Loops - MODE 3	R79 (A61)	R79 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A61 (authorize change in steam generator level requirement from greater than or equal to 6% to greater than or 32% following SG replacement). This change will NOT be incorporated into the Unit 2 TS Bases at this time since Unit 2 will utilize the original steam generators. NRC approved A61 via letter dated 05/05/2006.

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.4.6 RCS Loops - MODE 4	R79 (A61)	R79 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A61 (authorize change in steam generator level requirement from greater than or equal to 6% to greater than or 32% following SG replacement). This change will NOT be incorporated into the Unit 2 TS Bases at this time since Unit 2 will utilize the original steam generators. NRC approved A61 via letter dated 05/05/2006.
TS Bases 3.4.7 RCS Loops - MODE 5, Loops Filled	R79 (A61)	R79 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A61 (authorize change in steam generator level requirement from greater than or equal to 6% to greater than or 32% following SG replacement). This change will NOT be incorporated into the Unit 2 TS Bases at this time since Unit 2 will utilize the original steam generators. NRC approved A61 via letter dated 05/05/2006.
TS Bases 3.4.10 Pressurizer Safety Valves	R89 (A66)	R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.
TS Bases 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)	R89 (A66)	R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.
TS Bases 3.4.12 Cold Overpressure Mitigation System (COMS)	R89 (A66)	R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.4.13 RCS Operational LEAKAGE	R47 (A38) ---------- R68 (A56)	R47 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A38 (incorporate voltage-based Alternate Repair Criteria for steam generator tubes). This change was NOT incorporated into the Unit 2 TS Bases. Changes to TS Bases 3.4.13 per R47 were revised in entirety by R82; thus, no markup is included. NRC approved A38 via letter dated 02/26/2002. ---------- R68 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A56 (revise the Updated FSAR by modifying the design and licensing basis to increase the postulated primary-to-secondary leakage in the faulted steam generator following a main steamline break accident from 1 to 3 gallons per minute). This change was NOT incorporated into the Unit 2 TS Bases. Changes to TS Bases 3.4.13 per R68 were revised in entirety by R82; thus, no markup is included. NRC approved A56 via letter dated 03/10/2005.
TS Bases 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage	R89 (A66)	R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.5.2 ECCS - Operating	R14 ---------- R54 (A43) ---------- R57 (A48) ---------- R89 (A66)	R14 revised the Unit 1 TS Bases to update the hotleg recirculation timeframe. Changes per R14 were SUPERSEDED by R61, and were NOT applied to Unit 2. Since the changes per R14 were SUPERSEDED, no markup is provided. R14 was provided to NRC via letter dated 02/09/1998. ---------- R54 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A43 (revise, for ONE TIME ONLY, a portion of SR 3.5.2.3). This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS Bases. NRC approved A43 via letter dated 05/01/2003. ---------- R57 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A48. The change to the TS Bases updated the hotleg recirculation timeframe. Changes per R57 (A48) were SUPERSEDED by R61, and were NOT applied to Unit 2. Since the changes per R57 (A48) revision were SUPERSEDED, no markup is provided. A48 was provided to NRC via letter dated 10/08/2003. ---------- R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.
TS Bases 3.5.4 Refueling Water Storage Tank (RWST)	R61 (A40/A48) ---------- R88 (A67)	R61 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A40 and A48 (specified maximum number of TPBARs that can be irradiated in the Unit 1 reactor core). TPBARS will NOT be used on Unit 2; however, the boron concentration changes per R61 will be used on Unit 2. NRC approved Amendments 40 and 48 via letters dated 09/23/2002 and 10/08/2003, respectively. ********** Since the Unit 2 TS markup is based on R88, no markup is required for R61. ---------- R88 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A67 (revise the maximum number of TPBARs that can be irradiated in the Unit 1 reactor core to 400.) TPBARS will NOT be used on Unit 2; however, the boron concentration changes per R88 (A67) were incorporated herein. NRC approved A67 via letter dated 01/18/2008.

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.6.4 Containment Pressure	R71 (A59)	R71 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A59 (support steam generator replacement by allowing TEMPORARY use of penetrations in Shield Building Dome during Modes 1-4). This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS Bases. NRC approved A59 via letter dated 01/06/2006.
TS Bases 3.6.6 Containment Spray System	R89 (A66)	R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.
TS Bases 3.6.8 Hydrogen Mitigation System (HMS)	R16 (A10)	R16 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A10 (ONE TIME EXCEPTION to allow certain hydrogen igniters to be inoperable for a LIMITED TIME PERIOD). This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS Bases. NRC approved A10 via letter dated 06/09/1998.
TS Bases 3.6.9 Emergency Gas Treatment System (EGTS)	R71 (A59)	R71 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A59 (support steam generator replacement by allowing TEMPORARY use of penetrations in Shield Building Dome during Modes 1-4). Developmental Revision A for the Unit 2 TS stated, "This change will not be applied to Unit 2." NRC approved A59 via letter dated 01/06/2006. ********** DISCREPANCY: Contrary to the above, a minor portion of R71 (i.e., "See TS Bases 3.6.15, Shield Building, for additional information on EGTS.") was incorporated into Developmental Revision A of the Unit 2 TS Bases. Since this wording is nothing more than an aid to indicate where additional information on EGTS can be found, leaving this statement in the TS Bases has no impact on the Bases. The wording will be left in the TS Bases.
TS Bases 3.6.11 Ice Bed	R81 (A62)	R81 (A62) revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A62 (support steam generator replacement - increased minimum ice weight and total weight of stored ice.) Developmental Revision D to the Unit 2 TS and TS Bases (submitted to the NRC on 10/12/2010) revised the ice weights to be consistent with Unit 1.

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.6.12 Ice Condenser Doors	R06 (A03)	R06 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A03 (allow A ONE TIME EXTENSION of the 3 month surveillance requirement for the ice condenser lower inlet doors). This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS Bases. NRC approved A03 via letter date 09/09/1996.
TS Bases 3.6.15 Shield Building	R71 (A59)	R71 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A59 (support steam generator replacement by allowing TEMPORARY use of penetrations in Shield Building Dome during Modes 1-4). This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS Bases. NRC approved A59 via letter dated 01/06/2006.
TS Bases 3.7.1 Main Steam Safety Valves (MSSVs)	R41 (A31) ---------- R89 (A66)	R41 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by A31 {approve Power Uprate using Leading Edge Flow Meter (LEFM) for Unit 1}. This change will NOT be incorporated into the Unit 2 TS at this time. NRC approved A31 via letter dated 01/19/2001. ---------- R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.
TS Bases 3.7.2 Main Steam Isolation Valves (MSIVs)	R89 (A66)	R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves	R76 ---------- R89 (A66)	R76 revised the Unit 1 TS Bases (the change per R76 that is applicable to this TS Bases portion was the elimination of feedwater tempering flow; this was part of steam generator replacement). This change is NOT applicable to Unit 2; it was NOT incorporated into the Unit 2 TS Bases. R76 was provided to NRC via letter dated 05/23/2007. ---------- R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.
TS Bases 3.7.5 Auxiliary Feedwater (AFW) System	R89 (A66)	R89 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A66 (replace references to ASME Section XI with the ASME Operation and Maintenance Code for IST activities, and remove reference to "applicable supports" from the IST program. R89 also changed the applicability of SR 3.0.2 provisions to other normal and accelerated frequencies specified as two years or less in the IST program.). The changes per R89 were incorporated herein. NRC approved A66 via letter dated 12/18/2006.
TS Bases 3.7.6 Condensate Storage Tank (CST)	R41 (A31)	R41 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by A31 {approve Power Uprate using Leading Edge Flow Meter (LEFM) for Unit 1}. This change will NOT be incorporated into the Unit 2 TS Bases at this time. NRC approved A31 via letter dated 01/19/2001.
TS Bases 3.7.10 Control Room Emergency Ventilation System (CREVS)	R91 (A70)	R91 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A70 (Adopt TSTF Change Traveler TSTF-448, R3: Control Room Envelope Habitability.). The changes per R91 were incorporated herein. NRC approved A70 via letter dated 10/08/2008.
TS Bases 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)	R87	R87 revised the Unit 1 TS Bases because of DCN 52220-A: tied the ABI and CVI signals together so that either signal initiates the other signal. The changes per R87 were incorporated herein. R87 was provided to the NRC via letter dated 09/22/2008.
TS Bases 3.7.14 Secondary Specific Activity	R47 (A38)	R47 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A38 (incorporate voltage-based Alternate Repair Criteria for steam generator tubes). This change will NOT be incorporated into the Unit 2 TS Bases at this time. NRC approved A38 via letter dated 02/26/2002.

TS BASES SUBSECTION / TITLE	WBN UNIT 1 APPROVAL	COMMENTS
TS Bases 3.8.1 AC Sources - Operating	R50 (A39)	R50 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS A39 (revise LCO 3.8.1's allowed outage time to restore an inoperable emergency diesel generator to operable status from 72 hours to 14 days). Developmental Revision A for the Unit 2 TS Bases stated, "This change will not be applied to Unit 2." NRC approved A39 via letter dated 07/01/2002. ********** DISCREPANCY: Contrary to the above, the changes incorporated version of TS Bases 3.8.1 that was provided in Developmental Revision A of the Unit 2 TS Bases included the following verbiage at the end of SR 3.8.1.14: "Prior to performance of this SR in Modes 1 or 2, actions are taken to establish that adequate conditions exist for performance of the SR. The required actions are defined in Bases Table 3.8.1-2." This verbiage was not shown in the markup provided for Developmental Revision A of the Unit 2 TS Bases, and the verbiage should NOT have been incorporated. This wording will be removed from the Unit 2 TS Bases via a future Developmental Revision to the Unit 2 TS Bases.
TS Bases 3.9.8 Reactor Building Purge Air Cleanup Units	R87	R87 revised the Unit 1 TS Bases because of DCN 52220-A: tied the ABI and CVI signals together so that either signal initiates the other signal. The changes per R87 were incorporated herein. R87 was provided to the NRC via letter dated 09/22/2008.

Bases

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Pressurizer Water Level-High (continued)

reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level-High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

TSTF-169, R1

DELETE

10. Reactor Coolant Flow-Low

~~a. Reactor Coolant Flow-Low (Single Loop)~~

The Reactor Coolant Flow-Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

TSTF-169, R1

REPLACE

P-7.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE in MODE 1 above P-8.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions

TSTF-169, R1

INSERT

Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled.

(continued)

R90

TSTF-169, R1
INSERT
and above the P-7 setpoint,

TSTF-169, R1
DELETE

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Reactor Coolant Flow-Low ~~(Single Loop)~~

in the core. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip ~~(Function 10.b)~~ because of the lower power ~~level and the~~ greater margin to the design limit DNBR.

TSTF-169, R1
REPLACE WITH
10.

The Reactor Coolant Flow-Low Trip Setpoint and Allowable Value are specified in % indicated loop flow, however, the Eagle-21™ values entered through the MMI are specified in an equivalent % differential pressure.

TSTF-169, R1
INSERT
because of the higher power level.

b. Reactor Coolant Flow-Low (Two Loops)

The Reactor Coolant Flow-Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow-Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

TSTF-169, R1
DELETE

TSTF-169, R1
INSERT
Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since there is insufficient heat production to generate DNB conditions.

R90

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~b. Reactor Coolant Flow-Low (Two Loops) (continued)~~

~~The Reactor Coolant Flow-Low Trip Setpoint and Allowable Value are specified in % indicated loop flow, however, the Eagle 21™ values entered through the MMI are specified in an equivalent % differential pressure.~~

TSTF-169, R1

DELETE

11. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint, a loss of voltage detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low ~~(Two Loops)~~ Trip Setpoint is reached. The loss of voltage in two loops must be sustained for a length of time equal to or greater than that set in the time delay. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

TSTF-169, R1

INSERT

in two or more RCS loops.

The LCO requires one Undervoltage RCP channel per bus to be OPERABLE.

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

12. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps.

(continued)

R90

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

12. Underfrequency Reactor Coolant Pumps (continued)

thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected on two or more RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow – Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

TSTF-169, R1

DELETE

The LCO requires one Underfrequency RCP channel per bus to be OPERABLE.

TSTF-169, R1

INSERT

in two or more RCS loops.

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

13. Steam Generator Water Level-Low Low

Loss of the steam generator as a heat sink can be caused by the loss of normal feedwater, a station blackout or a feedline rupture. Feedline ruptures inside containment are protected by the containment high pressure trip Function, based on a 1994 TVA analysis (Ref. 3). Feedline ruptures outside containment and the other causes of the heat sink loss are protected by the SG Water Level Low-Low trip Function.

The SG Water Level – Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

16. Reactor Trip System Interlocks (continued)

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip may be blocked, and this Function would no longer be necessary. In MODE 3, 4, 5, or 6, the P-6 interlock is not required to be OPERABLE because the NIS Source Range is providing core protection.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure–Low;
- Pressurizer Water Level–High;
- Reactor Coolant Flow–Low (Two Loops)
- Undervoltage RCPs; and
- Underfrequency RCPs.

TSTF-169, R1

REPLACE WITH

(in two or more RCS loops)

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure–Low;
- Pressurizer Water Level–High;

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Low Power Reactor Trips Block, P-7 (continued)

Reactor Coolant Flow-Low (Two Loops);

Undervoltage RCPs; and

Underfrequency RCPs.

TSTF-169, R1

REPLACE WITH

(in two or more RCS loops)

Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS.

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1.

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint.

In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8

TSTF-169, R1

DELETE

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 48% power as determined by two-out-of-four NIS power range detectors. Above approximately 48% power the P-8 interlock automatically enables the Reactor Coolant Flow-Low ~~(Single Loop)~~ reactor trip on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 48% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

R90

(continued)

BASES

ACTIONS (continued)

D.1.1, D.1.2, D.2.1, D.2.2, and D.3

Condition D applies to the Power Range Neutron Flux-High Function.

The NIS power range detectors provide input to the CRD System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 7).

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to ≤ 75% RTP within 12 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one of the NIS power range detectors inoperable, 1/4 of the radial power distribution monitoring capability is lost.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within 6 hours and the QPTR monitored once every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels ≥ 75% RTP. The 6 hour Completion Time and the 12 hour Frequency are consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

As an alternative to the above actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also

TSTF-418, R2
REPLACE WITH
72

TSTF-418, R2
REPLACE WITH
Reference 14

TSTF-418, R2
REPLACE WITH
78

TSTF-418, R2
REPLACE WITH
72

TSTF-418, R2
DELETE

TSTF-418, R2
REPLACE WITH
Seventy-eight

TSTF-418, R2
INSERT
The 78-hour Completion Time includes 72 hours for channel corrective maintenance and an additional 6 hours for the MODE reduction as required by Required Action D.3.

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
is

(continued)

R90

BASES

ACTIONS

D.1.1, D.1.2, D.2.1, D.2.2, and D (continued)

TSTF-418, R2
REPLACE WITH
12

allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 7.

TSTF-418, R2
REPLACE WITH
Reference 14

Required Action D.2.2 has been modified by a Note which only requires SR 3.2.4.2 to be performed if the Power Range Neutron Flux input to QPTR becomes inoperable. Failure of a component in the Power Range Neutron Flux channel which renders the High Flux trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.

E.1 and E.2

TSTF-418, R2
REPLACE WITH
72

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux-Low; and
- Power Range Neutron Flux-High Positive Rate

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

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

TSTF-418, R2
REPLACE WITH
Reference 14

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

TSTF-418, R2
REPLACE WITH
12

(continued)

BASES

TSTF-418, R2
REPLACE WITH
14

TSTF-418, R2
REPLACE WITH
72

TSTF-418, R2
TSTF-169, R1
REPLACE WITH
Placing the channel in the tripped condition when above the P-8 setpoint results in a partial trip condition requiring only one additional channel in the same loop to initiate a reactor trip. Two tripped channels in each of two RCS loops are required to initiate a reactor trip when below the P-8 setpoint and above the P-7 setpoint. This Function does not have to be OPERABLE below the P-7 setpoint because there is no loss of flow trip below the P-7 setpoint. There is insufficient heat production to generate DNB conditions below the P-7 setpoint. The 72 hours allowed to place the channel in the tripped condition is justified in Reference 14. An additional 6 hours is allowed to reduce THERMAL Power to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

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

M.1 and M.2

Condition M applies to the following reactor trip Functions:

- Undervoltage RCPs; and
- Underfrequency RCPs.

TSTF-418, R2
REPLACE WITH
12

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

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

TSTF-169, R1
DELETE

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

N.1 and N.2

Condition N applies to the Reactor Coolant Flow—Low (Single Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be placed in trip within 6 hours. If the channel cannot be restored to OPERABLE status or the channel placed in trip within the 6 hours, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the unit in a MODE where the LCO is no longer applicable. This trip Function does not have to be OPERABLE below the P-8 setpoint because

TSTF-418, R2
REPLACE WITH
72

(continued)

R90

BASES

ACTIONS

N.1 and N.2 (continued)

TSTF-169, R1
DELETE

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

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
Reference 14

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

O.1 and O.2

TSTF-418, R2
DELETE

TSTF-418, R2
REPLACE WITH
Placing the channel

TSTF-418, R2
REPLACE WITH
72

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

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

P.1 and P.2

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

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
24

TSTF-418, R2
REPLACE WITH
Reference 14

(continued)

REO

BASES

TSTF-418, R2

REPLACE WITH

24

TSTF-418, R2

INSERT

The 24 hours allowed to restore the inoperable RTS Automatic Trip Logic train to OPERABLE status is justified in Reference 14.

P.1 and P.2 (continued)

to restore the train to OPERABLE status (Required Action P.1) or the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours (Required Action P.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action P.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

TSTF-411, R1

REPLACE WITH

24 hours are allowed for train corrective maintenance

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

Q.1 and Q.2

Condition Q applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1 hour and 6 hour Completion Times are equal to the time allowed LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the plant in MODE 3 removes the requirement for this particular Function.

TSTF-411, R1

INSERT

The 24 hour Completion Time is justified in Reference 15.

TSTF-411, R1

REPLACE WITH

Placing the unit in Mode 3 results in Condition C entry while RTB(s) are inoperable.

The Required Actions have been modified by two Notes. Note 1 allows one channel to be bypassed for up to 2 hours for surveillance testing, provided the other channel is OPERABLE. Note 2 allows one RTB to be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE. The 2 hour time limit is justified in Reference 7.

The Required Actions have been modified by a Note. The Note allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. The 4-hour time limit is justified in Reference 15.

R.1 and R.2

Condition R applies to the P-6 and P-10 interlocks. With one channel inoperable for one-out-of-two or two-out-of-four

(continued)

R90

BASES

ACTIONS

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

The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. With the RTBs open and the plant in MODE 3, this trip Function is no longer required to be OPERABLE. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 2 hours for the reasons stated under Condition Q.

The Completion Time of 48 hours for Required Action T.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

TSTF-418, R2

REPLACE WITH

72

U.1.1, U.1.2, and U.2

Condition U applies to the Steam Generator Water Level--Low-Low reactor trip Function.

A known inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition requires only one out of two logic for actuation of the two out of three trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If a channel fails, it is placed in the tripped condition and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD time delay by adjustment of the single steam generator time delay calculation (T_S) to match the multiple steam generator time delay calculation (T_M) for the affected protection set, through the Man Machine Interface.

TSTF-418, R2

REPLACE WITH

14.

If the inoperable channel cannot be restored or placed in the tripped condition within the specified Completion Time, the plant must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to

(continued)

R90

BASES

ACTIONS

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

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

TSTF-418, R2
REPLACE WITH
14

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

V.1 and V.2

TSTF-418, R2
REPLACE WITH
12

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

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

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

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
14

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

(continued)

R90

BASES

ACTIONS (continued)

W.1 and W.2

Condition W applies to the following reactor trip functions:

- Overtemperature ΔT;
- Overpower ΔT; and
- Pressurizer Pressure-High

TSTF-418, R2
REPLACE WITH
72

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

TSTF-418, R2
REPLACE WITH
14

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

TSTF-418, R2
REPLACE WITH
12

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

X.1 and X.2

TSTF-418, R2
REPLACE WITH
14

Condition X applies to the following reactor trip functions:

- Pressurize Pressure-Low;
- Pressurizes Water Level-High; and

(continued)

R90

TSTF-418, R2

DELETE

BASES

ACTIONS

X.1 and X.2 (continued)

~~• Reactor Coolant Flow-Low (Two Loops).~~

TSTF-418, R2

REPLACE WITH

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition when above the P-7 setpoint results in a partial trip condition requiring only one additional channel to initiate a reactor trip. These Functions do not have to be OPERABLE below the P-7 setpoint since there is insufficient heat production to generate DNB conditions below the P-7 setpoint. The 72 hours allowed to place the channel in the tripped condition is justified in Reference 14. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

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

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

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

Y.1 and Y.2

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

TSTF-418, R2

REPLACE WITH

12

TSTF-418, R2

REPLACE WITH

72

TSTF-418, R2

REPLACE WITH

14

TSTF-418, R2

REPLACE WITH

14

(continued)

R90

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%. Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 15% RTP and that 96 hours is allowed for performing the first Surveillance after reaching 15% RTP. This surveillance is typically performed at 50% RTP to ensure the results of the evaluation are more accurate and the adjustments more reliable. Ninety-six (96) hours are allowed to ensure Xenon stability and allow for instrumentation alignments.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

TSTF-411, R1

REPLACE WITH

62

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The bypass breaker test shall include a local shunt trip. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

(continued)

R90

TSTF-411, R1

REPLACE WITH

92

TSTF-411, R1

REPLACE WITH

62

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.4 (continued)

The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

TSTF-411, R1

REPLACE WITH

justified in Reference 15.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection Function. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

TSTF-411, R1

REPLACE WITH

92

SR 3.3.1.6

TSTF-411, R1

REPLACE WITH

justified in Reference 15.

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 6 days is allowed for performing the first surveillance after reaching 50% RTP.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

(continued)

R90

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days.

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

TSTF-411, R1
REPLACE WITH
184

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

TSTF-411, R1
REPLACE WITH
15

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

SR 3.3.1.7 is modified by a Note that this test shall include verification that the P-10 interlock is in the required state for the existing unit condition.

TSTF-411, R1
REPLACE WITH
184

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

SR 3.3.1.8

TSTF-411, R1
REPLACE WITH
References 9 and 15.

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by two Notes. Note 1 provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.8 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for greater than 4 hours, this Surveillance must be performed within 4 hours after entry into MODE 3. Note 2 states that this test shall include verification that the P-6 interlock is in the required state for the existing unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 31 days prior to reactor startup and 4 hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to

(continued)

R90

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.8 (continued)

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

DELETE

REPLACE WITH

source and intermediate range channels are OPERABLE channels

SR 3.3.1.9

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

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

(continued)

R90

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.11 (continued)

the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a COT of RTS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip, Reactor Trip from Manual SI, and the Reactor Trip from Automatic SI Input from ESFAS. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for these Reactor Trip Functions for the Reactor Trip Breakers. The test shall also verify OPERABILITY of the Reactor Trip Bypass Breakers for these Functions. Independent verification of the Reactor Trip Bypass Breakers undervoltage and shunt trip mechanisms is not required.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

TSTF-311

REPLACE WITH

exceeding the P-9 interlock whenever the unit has been in MODE 3. This

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification

(continued)

R90

BASES

SURVEILLANCE REQUIREMENTS

TSTF-311

REPLACE WITH

exceeding the P-9 interlock.

SR 3.3.1.14 (continued)

of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.15

SR 3.3.1.15 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in Technical Requirements Manual, Section 3.3.1 (Ref. 8). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

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

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

TSTF-411, R1

INSERT

(Ref. 11),

(continued)

R90

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 (continued)

TSTF-411, R1

INSERT

(Ref. 12),

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each channel's response must be verified every 18 months on a STAGGERED TEST BASIS. Testing of the final actuation devices is included in the testing. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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

(continued)

BASES

REFERENCES

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

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

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

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

TSTF-418, R2

INSERT

14. WCAP-14333 P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.

15. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.

3.3.1, "Reactor Trip System Response Times."

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

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

11. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996

12. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

13. WCAP-16067-P, Rev. 0, "RCS Flow Measurement Using Elbow Tap Methodology at Watts Bar Unit 1," April 2003.

INSERT

, Westinghouse Letter WAT-D-10128.

R90

BASES

ACTIONS

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

isolation, failure of one or both channels in one train renders the train inoperable. Condition B, therefore, encompasses both situations. 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. If the 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. The allowance of 48 hours is justified in Reference 7.

TSTF-418, R2

INSERT

The 24 hours allowed for restoring the inoperable train to OPERABLE status are justified in Reference 17.

C.1, C.2.1 and C.2.2

Condition C applies to the automatic actuation logic and actuation relays for the following functions:

- SI;
- Containment Spray;
- Phase A Isolation;
- Phase B Isolation; and
- Automatic Switchover to Containment Sump.

TSTF-418, R2

REPLACE WITH

30

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the 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 (12 hours total time) and in MODE 5

TSTF-418, R2

REPLACE WITH

24

(continued)

R90

BASES

ACTIONS

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

within an additional 30 hours (42 hours total time). The 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.

TSTF-418, R2
REPLACE WITH
60

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 7) that 4 hours is the average time required to perform channel surveillance.

TSTF-418, R2
REPLACE WITH
train

D.1, D.2.1, and D.2.2

Condition D applies to:

- Containment Pressure-High;
- Pressurizer Pressure-Low;
- Steam Line Pressure-Low; and
- Steam Line Pressure-Negative Rate-High.

TSTF-418, R2
REPLACE WITH
72

If one channel is inoperable, 6 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-three configuration that satisfies redundancy requirements.

TSTF-418, R2
INSERT
The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition are justified in Reference 17.

(continued)

TSTF-418, R2
REPLACE WITH
72

BASES

ACTIONS

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

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the plant be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

TSTF-418, R2
REPLACE WITH
12

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, these Functions are no longer required OPERABLE.

TSTF-418, R2
DELETE

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 7.

TSTF-418, R2
DELETE

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
17

E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Spray Containment Pressure-High High;
- Steam Line Isolation Containment Pressure-High High; and
- Containment Phase B Isolation Containment Pressure-High High.

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with

R90

(continued)

BASES

ACTIONS

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

two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray.

TSTF-418, R2

REPLACE WITH

72

To avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 6 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 6 hours, requires the plant be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, these Functions are no longer required OPERABLE.

REPLACE WITH

The Required Actions are modified by a Note that allows placing one channel in bypass for up to 12 hours while performing routine surveillance testing. The channel to be tested can be tested in bypass with the inoperable channel also in bypass. The time limit is justified in Reference 17.

~~The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 4 hours for surveillance testing. Placing a second channel in the bypass condition for up to 4 hours for testing purposes is acceptable based on the results of Reference 7.~~

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

R90

(continued)

BASES

ACTIONS

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

- 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.

TSTF-418, R2

INSERT

The 24 hours allowed for restoring the channel to OPERABLE status or to place it in the tripped condition are justified in Reference 17.

G.1, G.2.1 and G.2.2

Condition G applies to the automatic actuation logic and actuation relays for the Steam Line Isolation and AFW actuation Functions.

The action addresses the train orientation of the SSPS and the master and slave relays for these functions. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the plant must be brought to 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

TSTF-418, R2

REPLACE WITH

24

RGO

(continued)

BASES

ACTIONS

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

required plant conditions from full power conditions in an orderly manner and without challenging plant systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the plant does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 7) assumption that 4 hours is the average time required to perform channel surveillance.

TSTF-418, R2

INSERT

The 24 hours allowed for restoring the channel to OPERABLE status or to place it in the tripped condition are justified in Reference 17.

TSTF-418, R2

REPLACE WITH

24

H.1, H.2.1 and H.2.2

Condition H applies to the automatic actuation logic and actuation relays for the Turbine Trip and Feedwater Isolation Function.

This action addresses the train orientation of the SSPS and the master and slave relays for this Function. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status or the plant must be placed in MODE 3 within 6 hours and in MODE 4 in the following 6 hours. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. The allowed Completion Times are reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging plant systems. These Functions are no longer required in MODE 4. Placing the plant in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the plant does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 7)

R90

(continued)

BASES

ACTIONS

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

TSTF-418, R2

REPLACE WITH

72

assumption that 4 hours is the average time required to perform channel surveillance.

I.1, I.2.1 and I.2.2

TSTF-418, R2

REPLACE WITH

The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition are justified in Reference 17.

Condition I applies to SG Water Level - High High (P-14).

If one channel is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two logic will result in actuation. The 6 hour Completion Time is justified in Reference 7. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the plant to be placed in MODE 3 in 6 hours and in MODE 4 in the following 6 hours. The allowed Completion Times are rea on operating experience, to reach MODE 4 fro conditions in an orderly manner and without plant systems. In MODE 4, these Functions a required OPERABLE.

TSTF-418, R2

REPLACE WITH

72

TSTF-418, R2

REPLACE WITH

12

The Required Actions have been modified by a Note that allows placing an inoperable channel in bypassed condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows a channel to be placed in bypass for up to 4 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 6 hours allowed to place the inoperable channel in the tripped condition, and the 4 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference 7.

TSTF-418, R2

REPLACE WITH

The 12 hours allowed for testing are justified by Reference 17.

J.1 and J.2

Condition J applies to the AFW pump start on trip of all MFW pumps.

The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to

R90

(continued)

BASES

ACTIONS

J.1 and J.2 (continued)

an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the plant in MODE 3. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. In MODE 3, the plant does not have any analyzed transients or conditions that require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 7.

K.1, K.2.1 and K.2.2

Condition K applies to RWST Level – Low Coincident with Safety Injection and Coincident with Containment Sump Level – High.

TSTF-418, R2

REPLACE WITH

72

RWST Level – Low Coincident With SI and Coincident With Containment Sump Level – High provides actuation of switchover to the containment sump. Note that this Function requires the comparators to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in a premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two-out-of-three logic configuration, which satisfies the requirement to allow another failure without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The 6 hour Completion Time is justified in Reference 7. If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours, the plant must be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the plant

TSTF-418, R2

REPLACE WITH

References 10, 17, and 19.

R90

(continued)

BASES

ACTIONS

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

does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows placing a second channel in the bypass condition for up to 4 hours for surveillance testing. The total of 12 hours to reach MODE 3 and 4 hours for a second channel to be bypassed is acceptable based on the results of Reference 7.

REPLACE WITH

The Required Actions are modified by a Note that allows placing one channel in bypass for up to 12 hours while performing routine surveillance testing. The channel to be tested can be tested in bypass with the inoperable channel also in bypass. The time limit is justified in Reference 17.

L.1, L.2.1 and L.2.2

Condition L applies to the P-11 interlock.

With one channel inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing plant condition, 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 conditions in an orderly manner and without challenging plant systems. Placing the plant in MODE 4 removes all requirements for OPERABILITY of these interlocks.

R90

(continued)

BASES

TSTF-418, R2
REPLACE WITH
72

ACTIONS (continued)

M.1.1, M.1.2 and M.2

Condition M is applicable to the SG Water Level Low-Low Function.

TSTF-418, R2
REPLACE WITH
to restore the channel to OPERABLE status or to place it

A known channel inoperable, must be restored to OPERABLE status, or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one out of two logic for actuation of the two out of three trip. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

TSTF-418, R2
REPLACE WITH
are

If a channel fails, it is placed in the tripped condition and does not affect the TTD setpoint calculations for the remaining OPERABLE channels. It is then necessary for the operator to force the use of the shorter TTD Time Delay by adjustment of the single SG time delay calculation (T_S) to match the multiple SG time delay calculation (T_M) for the affected protection set, through the Man-Machine Interface.

TSTF-418, R2
REPLACE WITH
17

If the inoperable channel cannot be restored or placed in the tripped condition within the specified Completion Time, the plant must be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to place the plant in MODE 3 from MODE 1 full power conditions in an orderly manner and without challenging plant systems.

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

TSTF-418, R2
REPLACE WITH
12

TSTF-418, R2
REPLACE WITH
17

(continued)

R90

BASES

ACTIONS (continued)

N.1 and N.2

Condition N applies to the Vessel ΔT Equivalent to Power Function.

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

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

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

TSTF-418, R2
REPLACE WITH
12

O.1 and O.2

TSTF-418, R2
REPLACE WITH
17

Condition O applies to North or South MSVV Room Water Level - High.

If one channel is inoperable, 6 hours are allowed to restore that channel to OPERABLE status or place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two logic will result in actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 11.

TSTF-418, R2
REPLACE WITH
72

TSTF-418, R2
REPLACE WITH
References 10 and 17

REV

(continued)

BASES

TSTF-418, R2

REPLACE WITH

72

ACTIONS

0.1 and 0.2 (continued)

INSERT

hours

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the plant to be placed in MODE 3 within the following 6 hours. The allowed Completion Time of 6 is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. In MODE 3, these functions are no longer required OPERABLE.

TSTF-418, R2

REPLACE WITH

12

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 11.

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

TSTF-418, R2

REPLACE WITH

References 10 and 17.

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

R90

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

SR 3.3.2.1

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

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

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

TSTF-411, R1

REPLACE WITH

92

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives,

R90

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.2 (continued)

TSTF-411, R1

REPLACE WITH

92

are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

TSTF-411, R1

REPLACE WITH

justified in Reference 18.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

TSTF-411, R1

REPLACE WITH

92

SR 3.3.2.4

SR 3.3.2.4 is the performance of a COT.

TSTF-411, R1

REPLACE WITH

The Frequency of 92 days is justified in Reference 18.

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

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

TSTF-411, R1

REPLACE WITH

Reference 6.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis (Ref. 7) when applicable, and the setpoint methodology (Ref. 6).

R90

(continued)

TSTF-411, R1

REPLACE WITH

184

TSTF-411, R1

REPLACE WITH

18

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.4 (continued)

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

TSTF-411, R1

REPLACE WITH

References 10 and 18.

SR 3.3.2.5

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

For ESFAS slave relays which are Westinghouse type AR relays, the SLAVE RELAY TEST is performed every 18 months. The frequency is based on the relay reliability assessment presented in Reference 13. This reliability assessment is relay specific and applies only to Westinghouse type AR relays with AC coils. Note that, for normally energized applications, the relays may require periodic replacement in accordance with the guidance given in Reference 13.

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

SR 3.3.2.6

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

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

R98

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.10 (continued)

tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

TSTF-411, R1

INSERT

(Reference 15),

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

TSTF-411, R1

INSERT

(Reference 16),

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

Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

R90

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.10 (continued)

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

There is an additional note pertaining to this SR on Page 3 of Table 3.3.2-1 of the Technical Specification, which states the following (Ref. 14):

~~Note h: For the time period between February 23, 2000 and prior to turbine restart (following the next time the turbine is removed from service), the response time test requirement of SR 3.3.2.10 is not applicable for 1-FSV-47-027.~~

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per RTB cycle. This Frequency is based on operating experience demonstrating that undetected failure of the P-4 interlock sometimes occurs when the RTB is cycled.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

REFERENCES

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

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

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

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

(continued)

Not on K2

BASES

REFERENCES (continued)

5. Code of Federal Regulations, Title 10, Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

6. WCAP-12096, Rev. 7, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," March 1997.

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

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

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

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

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

12. Design Change Notice W-38238 associated documentation.

13. WCAP-13877, Rev. 1, "Reliability Assessment of Westinghouse Type AR Relays Used As SSPS Slave Relays," August 1998.

14. TVA's Letter to NRC dated February 25, 2000, "WBN Unit 1 Request for TS Amendment for TS 3.3.2 - ESFAS Instrumentation."

15. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

16. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

TSTF-411, R1; TSTF-418, R2

INSERT

17. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.

18. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.

19. Westinghouse letter to TVA, WAT-D-11248, "Revised Justification for Applicability of Instrumentation Technical Specification Improvements to the Automatic Switchover to Containment Sump Signal," June 2004

INSERT

, Westinghouse letter to TVA WAT-D-10128

R90

BASES

REFERENCES (continued)

5. Code of Federal Regulations, Title 10, Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

6. WCAP-12096, Rev. 7, "Westinghouse Setpoint Methodology for Protection System, Watts Bar 1 and 2," March 1997.

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

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

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

10. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar, Westinghouse letter to TVA WAT-D-10128.

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

12. Design Change Notice W-38238 associated documentation.

13. WCAP-13877, Rev. 1, "Reliability Assessment of Westinghouse Type AR Relays Used As SSPS Slave Relays," August 1998.

14. TVA's Letter to NRC dated February 25, 2000, "WBN Unit 1 Request for TS Amendment for TS 3.3.2 - ESFAS Instrumentation."

15. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

16. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

17. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.

18. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.

19. Westinghouse letter to TVA, WAT-D-11248, "Revised Justification for Applicability of Instrumentation Technical Specification Improvements to the Automatic Switchover to Containment Sump Signal," June 2004.

~~20. Letter from John G. Lamb (NRC) to Mr. Preston D. Swafford (TVA) dated March 4, 2009, Includes Enclosures (a) Amendment No. 75 to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant, Unit 1 and (b) NRC Safety Evaluation (SE) for Amendment No. 75.~~

Not on U2

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Vent Isolation Instrumentation

BASES

BACKGROUND

Containment Vent Isolation Instrumentation closes the containment isolation valves in the Containment Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Reactor Building Purge System may be in use during reactor operation and with the reactor shutdown.

Containment vent isolation is initiated by a safety injection (SI) signal or by manual actuation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss initiation of SI signals.

Redundant and independent gaseous radioactivity monitors measure the radioactivity levels of the containment purge exhaust, each of which will initiate its associated train of automatic Containment Vent Isolation upon detection of high gaseous radioactivity.

The Reactor Building Purge System has inner and outer containment isolation valves in its supply and exhaust ducts. This system is described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

INSERT 1

APPLICABLE SAFETY ANALYSES

The containment isolation valves for the Reactor Building Purge System close within six seconds following the DBA. The containment vent isolation radiation monitors act as backup to the SI signal to ensure closing of the purge air system supply and exhaust valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The Containment Vent Isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

INSERT 2

(continued)

R87

BASES

LCO (continued)

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Vent Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups and sample pump operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

Only the Allowable Value is specified for the Containment Purge Exhaust Radiation Monitors in the LCO. The Allowable Value is based on expected concentrations for a small break LOCA, which is more restrictive than 10 CFR 100 limits. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.

4. Safety Injection (SI)

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Safety Injection, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release significant fission product radioactivity into containment. Therefore, the Containment Vent Isolation Instrumentation must be OPERABLE in these MODES. SEE ADDITIONAL DISCUSSION IN THE BACKGROUND AND APPLICABLE SAFETY ANALYSIS SECTIONS

(continued)

R87

INSERT 1

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130, and -131 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. These changes will require that the containment ventilation instrumentation remain operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc are open to the Auxiliary Building ABSCE spaces.

R87

INSERT 2

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI generated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable. This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of a FHA in the Auxiliary Building.

R87

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

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

TSTF-411, R1

REPLACE WITH

justified in Reference 4.

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

TSTF-411, R1

REPLACE WITH

92

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

TSTF-411, R1

INSERT

The SR is modified by a Note stating that the surveillance is only applicable to the actuation logic of the ESFAS instrumentation.

SR 3.3.6.3

SR 3.3.6.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

TSTF-411, R1

INSERT

The SR is modified by a Note stating that the surveillance is only applicable to the actuation logic of the ESFAS instrumentation.

TSTF-411, R1

REPLACE WITH

justified in Reference 4.

TSTF-411, R1

REPLACE WITH

92

(continued)

R90

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

For these tests, the relay trip setpoints are verified and adjusted as necessary. The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.6.7

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

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

2. NUREG-1366, "Improvement to Technical Specification Surveillance Requirements," December 1992.

3. WCAP-13877, Rev. 1. "Reliability Assessment of Westinghouse Type AR Relays Used as SSPS Slave Relays." August 1998.

INSERT

4. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.

R90

B 3.3 INSTRUMENTATION

B 3.3.8 Auxiliary Building Gas Treatment (ABGTS) Actuation Instrumentation

BASES

BACKGROUND

The ABGTS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident or a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.12, "Auxiliary Building Gas Treatment System." The system initiates filtered exhaust of air from the fuel handling area, ECCS pump rooms, and penetration rooms automatically following receipt of a fuel pool area high radiation signal or a Containment Phase A Isolation signal. Initiation may also be performed manually as needed from the main control room.

High area radiation, monitored by either of two monitors, provides ABGTS initiation. Each ABGTS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. High radiation detected by any monitor or a Phase A isolation signal from the Engineered Safety Features Actuation System (ESFAS) initiates auxiliary building isolation and starts the ABGTS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the Auxiliary Building Secondary Containment Enclosure (ABSCE).

INSERT 3

APPLICABLE SAFETY ANALYSES

The ABGTS ensures that radioactive materials in the ABSCE atmosphere following a fuel handling accident or a LOCA are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the auxiliary building exhaust following a LOCA or fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

INSERT 4

The ABGTS Actuation Instrumentation satisfies Criterion 3 of the NRC Policy Statement.

(continued)

R87

BASES

LCO (continued)

3. Containment Phase A Isolation

Refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements.

APPLICABILITY

The manual ABGTS initiation must be OPERABLE in MODES 1, 2, 3, and 4 and when moving irradiated fuel assemblies in the fuel handling area, to ensure the ABGTS operates to remove fission products associated with leakage after a LOCA or a fuel handling accident. The Phase A ABGTS Actuation is also required in MODES 1, 2, 3, and 4 to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage.

High radiation initiation of the ABGTS must be OPERABLE in any MODE during movement of irradiated fuel assemblies in the fuel handling area to ensure automatic initiation of the ABGTS when the potential for a fuel handling accident exists. SEE ADDITIONAL DISCUSSIONS IN THE BACKGROUND AND APPLICABLE SAFETY ANALYSIS SECTIONS.

While in MODES 5 and 6 without fuel handling in progress, the ABGTS instrumentation need not be OPERABLE since a fuel handling accident cannot occur.

ACTIONS

The most common cause of channel inoperability is outright failure or drift sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

R87

INSERT 3

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment and/or annulus open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130, and -131 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. These changes will require that the containment ventilation instrumentation remain operable when moving irradiated fuel in the Auxiliary Building if the containment and/or annulus air locks, penetrations, equipment hatch, etc are open to the Auxiliary Building ABSCE spaces.

R87

INSERT 4

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI generated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable. This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of a FHA in the Auxiliary Building.

R87

BASES

ACTIONS (continued)

D.1, D.2, and D.3

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

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

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

~~SR 3.4.5.2~~

Was "6%"

SR 3.4.5.2 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) for required RCS loops. 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 for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

(continued)

Not on U2

BASES

LCO (continued)

The Note requires that the secondary side water temperature ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ 350°F. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. "3.4.6.2" was

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.

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).

Not on U2

(continued)

Was "does not account"

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

Was "≥ 6%"

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 ~~accounts~~ 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.

Was "< 6%"

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. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

Not on U2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels ~~greater than or equal to 32%~~ narrow range to provide an alternate method for decay heat removal.

Was "above 6%"

(continued)

Not on U2

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

LCO

WAS "≥6%"

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ~~greater than or equal to 32%~~ narrow range. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ~~greater than or equal to 32%~~ narrow range. Should the operating RHR loop fail, the SGs could be used to remove the decay heat.

Note 1 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 2 requires that the secondary side water temperature of each SG be less than or equal to 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature less than or equal to 350°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

(continued)

Not on U2

BASES

LCO (continued)

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink when it has an adequate water level and is OPERABLE.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ~~greater than or equal to 32%~~ narrow range.

Was "≥ 6%"

Operation in other MODES is covered by:

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

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels ~~less than 32%~~ narrow range redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

Was "< 6%"

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all

(continued)

Not on UL

BASES

ACTIONS

B.1 and B.2 (continued)

operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

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

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ~~greater than or equal to 32%~~ (value does not account for instrument error, Ref. 1) narrow range ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. 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.

WBS "≥ 6%"

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side water level is ~~greater than or equal to 32%~~ narrow range in at least two SGs,

(continued)

Not on U2

BASES

ACTIONS

A.1 (continued)

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

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below 350°F, overpressure protection is provided by the COMS System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

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

The pressurizer safety valve setpoint is ± 3% for OPERABILITY, however, the valves are reset to ± 1% during the surveillance to allow for drift.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, NB 7000, 1971 Edition through Summer 1973.

(continued)

R89

BASES

REFERENCES (continued)

2. Watts Bar FSAR, Section 15.0, "Safety Analyses."

3. WCAP-7769, Rev. 1, "Topical Report on Overpressure Protection for Westinghouse Pressurized Water Reactors," June 1972.

4. ~~ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."~~ **ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants"**

R89

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME OM Code, ~~Section XI~~ (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an inoperable PORV that is incapable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status.

The Note modifies this SR by stating that it is not required to be met with the block valve closed, in accordance with the Required Action of this LCO.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

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

2. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency."

3. **ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."** ~~**ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."**~~

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BASES

REFERENCES (continued)

7. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability, and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light Water Reactors,' pursuant to 10 CFR 50.44(f)."

8. **ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."** ~~**Boiler and Pressure Vessel Code, Section XI.**~~

9. Letter WAT-D-9448, "Tennessee Valley Authority Watts Bar Nuclear Plant Units 1 & 2 Revised COMS PORV Setpoints", August 27, 1994.

R89

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) OM Code, ~~Section XI~~ (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been

(continued)

R89

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Section 50.2, "Definitions--Reactor Coolant Pressure Boundary."

2. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (c), "Reactor Coolant Pressure Boundary."

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, Section V, "Reactor Containment," General Design Criterion 55, "Reactor Coolant Pressure Boundary Penetrating Containment."

4. U.S. Nuclear Regulatory Commission (NRC), "Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix V, WASH-1400 (NUREG-75/014), October 1975.

5. U.S. NRC, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," NUREG-0677, May 1980.

6. Watts Bar FSAR, Section 3.9, "Mechanical Systems and Components" (Table 3.9-17).

7. **ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."** ~~**Boiler and Pressure Vessel Code, Section XI.**~~

8. Title 10, Code of Federal Regulations, Part 50, Section 50.55a, "Codes and Standards," Subsection (g), "Inservice Inspection Requirements."

R89

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12-hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water by venting the ECCS pump casings and accessible suction and discharge piping high points ensures that the system will perform properly, injecting its full capacity into the RCS upon demand.* This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation. ~~A note is added to the FREQUENCY that surveillance performance is not required for safety injection hot leg injection lines until startup from the Fall 2003 Refueling Outage. (Ref. 7).~~

(continued)

Not on U2

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3 (continued)

*For the accessible locations, UT may be substituted to demonstrate the piping is full of water. An accessible ECCS high point is defined as one that:

1) Has a vent connection installed.
2) The high point can be vented with the dose received remaining within ALARA expectations. ALARA for venting ECCS high point vents is considered to not be within ALARA expectations when the planned, intended collective dose for the activity is unjustifiably higher than industry norm, or the licensee's past experience, for this (or similar) work activity.
3) The high point can be vented with industrial safety expectations remaining within the industry norm.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~Section XI of~~ the ASME OM Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pumps baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, ~~which encompasses Section XI~~ of the ASME Code. ~~Section XI of the~~ ASME OM Code provides the activities and Frequencies necessary to satisfy the requirements.

SR ~~3.5.2.5~~ and 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

(continued)

R89

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

Realignment of valves in the flow path on an SI signal is necessary for proper ECCS performance. These valves are secured in a throttled position for restricted flow to a ruptured cold leg, ensuring that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The advanced sump strainer design installed at WBN incorporates both the trash rack function and the screen function. Inspection of the advanced strainer constitutes fulfillment of the trash rack/screen inspection. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling System."

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plant."

3. Watts Bar FSAR, Section 6.3, "Emergency Core Cooling System."

4. FSAR Bar FSAR, Section 15.0, "Accident Analysis."

5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

6. IE Information Notice No. 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987.

~~7. WBN License Amendment Request WBN-TS-03-11 dated April 8, 2003.~~

Not on UL

BASES

APPLICABLE SAFETY ANALYSES (continued)

volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.

The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as 27 seconds, with offsite power available, or 37 seconds without offsite power.

~~Technical Specification Surveillance Requirements 3.5.1.4, "Accumulators," and 3.5.4.3, "RWST," match boron concentrations to the number of tritium producing burnable absorbers rods (TPBARs) installed in the reactor core. Watts Bar is authorized to place a maximum of 400 TPBARs into the reactor in an operating cycle. Generally, TPBARs act as burnable absorber rods normally found in similar reactor core designs. However, unlike burnable absorber rods which lose their poison effects over the life of the cycle, some residual effect remains in the TPBARs at the end of the cycle. When larger amounts of excess neutron poisons (as in the case with larger loads of TPBARs) are added to a core, there is competition for neutrons from all the poison and the negative worth of each poison (including the reactor coolant system (RCS) boron) decreases. The positive reactivity insertion due to the negative moderator coefficient that occurs during the cooldown from hot full power to cold conditions following a loss of coolant accident (LOCA) must be overcome by RCS boron. Because the RCS boron is worth less, it takes a higher concentration to maintain subcriticality.~~

For a large break LOCA Analysis, the minimum water volume limit of 370,000 gallons and the minimum boron concentration limit is used to compute the post LOCA sump boron concentration necessary to assure subcriticality. This

(continued)

Was chosen to be consistent with the minimum value specified for Unit 1.

BASES

APPLICABLE SAFETY ANALYSES (continued)

minimum value ~~depends on the number of TPBARs in the core as specified in the Core Operating Limits Report (COLR) for each operating cycle.~~ The large break LOCA is the limiting case since the safety analysis assumes least negative reactivity insertion.

The upper limit on boron concentration of 3300 ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 60°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The acceptable temperature range of 60°F to 105°F is assumed in the large break LOCA analysis, and the small break analysis value bounds the upper temperature limit of 105°F. The upper temperature limit of 105°F is also used in the containment OPERABILITY analysis. Exceeding the upper temperature limit will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water following a LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of the NRC Policy Statement.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

(continued)

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour. ~~When opening or closing Penetration 1-EQH-271-0010 or 1-EQH-271-0011 in the Shield Building Dome, the differential pressure between the Containment and the Annulus may exceed the equal to or greater than -0.1 and equal to or less than +0.3 psid requirement. During this operation, time is allowed for Containment/Annulus pressure equalization to be re-established.~~

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits (≥ -0.1 and ≤ +0.3 psid relative to the annulus, value does not account for instrument error, Ref. 3) ensures that plant operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. Watts Bar FSAR, Section 6.2.1, "Containment Functional Design."

2. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

3. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

Not on U2

BASES

ACTIONS

C.1 and C.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.6.6.1

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

SR 3.6.6.2

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

(continued)

R89

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criterion (GDC) 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal System," GDC 40, "Testing of Containment Heat Removal Systems, and GDC 50, "Containment Design Basis."

2. Watts Bar FSAR, Section 6.2, "Containment Systems."

3. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

4. ASME **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants,"** ~~**Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components,"**~~ American Society of Mechanical Engineers, New York.

R89

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.8.3 (continued)

experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

TEMPORARY CONDITION

LCO 3.6.8 is modified by Notes that provide temporary requirements for the HMS due to a condition discovered on April 3, 1998, wherein two Train A ignitors (30A and 31A) were found inoperable during surveillance testing. The ignitors are located in high radiation and temperature areas of Unit 1 containment and should be repaired with the reactor offline to avoid personnel safety hazards associated with making repairs online. The Notes are justified in Reference 4 on the basis the HMS will still be capable of performing its intended function. The Notes establish the following for the temporary period.

(1) This temporary specification will expire at WBN's next entry into MODE 3.

(2) The BASES of LCO 3.6.8 on page B3.6-51 is modified by defining that HMS Train A is considered OPERABLE with 32 of 34 ignitors OPERABLE. This allowance is only permitted for the condition where ignitors 30A and 31A are the only inoperable A-train ignitors.

(3) CONDITION B of LCO 3.618 is modified to allow two specific containment regions (Reactor Cavity Region and Steam Generator No. 4 Enclosure Lower Compartment Region) to have no OPERABLE ignitors for a period of up to 72 hours.

(4) SR 3.6.8.1 is modified to permit ≥ 32 ignitors energized for HMS Train A to demonstrate operability. The testing must be performed at an increased frequency of 46 days.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.44, "Standards for Combustible Gas Control Systems in Light Water-Cooled Power Reactors."

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

3. Watts Bar FSAR, Section 6.2.5A, "Hydrogen Mitigation System Description."

~~4. TVA letter to NRC from P. L. Pace, , "WBN Unit 1 - Request for TS Amendment for TS 3.6.8 - Hydrogen Mitigation System (HMS) (TS-98-011)," April 29, 1998.~~

Not on U2

B 3.6 CONTAINMENT SYSTEMS

B 3.6.11 Ice Bed

Was "2,158,000"

BASES

BACKGROUND

The ice bed consists of over ~~2,404,500~~ lbs of ice stored in 1944 baskets within the ice condenser. Its primary purpose is to provide a large heat sink in the event of a release of energy from a Design Basis Accident (DBA) in containment. The ice would absorb energy and limit containment peak pressure and temperature during the accident transient. Limiting the pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ice condenser is an annular compartment enclosing approximately 300° of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment, which, for normal plant operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal plant operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal plant operation. The upper plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets contain the ice within the ice condenser. The ice bed is considered to consist of the total volume from the bottom elevation of the ice baskets to the top elevation of the ice baskets. The ice baskets position the ice within the ice bed in an arrangement to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows

(continued)

Not on UL

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.2

The weighing program is designed to obtain a representative sample of the ice baskets. The representative sample shall include 6 baskets from each of the 24 ice condenser bays and shall consist of one basket from radial rows 1, 2, 4, 6, 8, and 9. If no basket from a designated row can be obtained for weighing, a basket from the same row of an adjacent bay shall be weighed.

The rows chosen include the rows nearest the inside and outside walls of the ice condenser (rows 1 and 2, and 8 and 9, respectively), where heat transfer into the ice condenser is most likely to influence melting or sublimation. Verifying the total weight of ice ensures that there is adequate ice to absorb the required amount of energy to mitigate the DBAs.

If a basket is found to contain less than ~~1237~~ lb of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The average weight of ice in these 21 baskets (the discrepant basket and the 20 additional baskets) shall be greater than or equal to ~~1237~~ lb at a 95% confidence level. [Value does not account for instrument error.]

Was "1110"

Weighing 20 additional baskets from the same bay in the event a Surveillance reveals that a single basket contains is less than ~~1237~~ lb ensures that no local zone exists that is grossly deficient in ice. Such a zone could experience early melt out during a DBA transient, creating a path for steam to pass through the ice bed without being condensed. The Frequency of 18 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 18 month Frequency, the weight requirements are maintained with no significant degradation between surveillances.

(continued)

Not on U2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.3

Verifying, by visual inspection, that the ice condenser inlet doors are not impaired by ice, frost, or debris provides assurance that the doors are free to open in the event of a DBA. For this unit, the Frequency of 18 months (3 months during the first year after receipt of license - the 3 month performances during the first year after receipt of license may be extended to coincide with plant outages) is based on door design, which does not allow water condensation to freeze, and operating experience, which indicates that the inlet doors very rarely fail to meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown. ~~The surveillance frequency is modified by a Note that permits a one-time extension until October 21, 1996 for performance of the three month surveillance whose due date (with 25 percent extension) falls on September 9, 1996. This provision allows performance of the surveillance to coincide with the plant mid-cycle outage and is justified by Reference 3.~~

SR 3.6.12.4

Verifying the opening torque of the inlet doors provides assurance that no doors have become stuck in the closed position. The value of 675 in-lb is based on the design opening pressure on the doors of 1.0 lb/ft^2. For this unit, the Frequency of 18 months (3 months during the first year after receipt of license - the 3 month performances during the first year after receipt of license may be extended to coincide with plant outages) is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no known factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which does not allow water condensation to freeze). Operating experience indicates that the inlet doors usually meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown. ~~The surveillance frequency is modified by a Note that permits a one-time extension until October 21, 1996, for performance of the three month surveillance whose due date (with 25 percent extension) falls on September 9, 1996. This provision allows performance of the surveillance to coincide with the plant mid-cycle outage and is justified by Reference 3.~~

(continued)

Not on U2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.5

The torque test Surveillance ensures that the inlet doors have not developed excessive friction and that the return springs are producing a door return torque within limits. The torque test consists of the following:

1. Verify that the torque, T(OPEN), required to cause opening motion at the 40° open position is ≤ 195 in-lb;

2. Verify that the torque, T(CLOSE), required to hold the door stationary (i.e., keep it from closing) at the 40° open position is ≥ 78 in-lb; and

3. Calculate the frictional torque, T(FRICT) = 0.5 {T(OPEN) - T(CLOSE)}, and verify that the T(FRICT) is ≤ 40 in-lb.

The purpose of the friction and return torque Specifications is to ensure that, in the event of a small break LOCA or SLB, all of the 24 door pairs open uniformly. This assures that, during the initial blowdown phase, the steam and water mixture entering the lower compartment does not pass through part of the ice condenser, depleting the ice there, while bypassing the ice in other bays. The Frequency of 18 months (3 months during the first year after receipt of license - the 3 month performances during the first year after receipt of license may be extended to concide with plant outages) is based on the passive nature of the closing mechanism (i.e., once adjusted, there are no known factors that would change the setting, except possibly a buildup of ice; ice buildup is not likely, however, because of the door design, which does not allow water condensation to freeze). Operating experience indicates that the inlet doors very rarely fail to meet their SR acceptance criteria. Because of high radiation in the vicinity of the inlet doors during power operation, this Surveillance is normally performed during a shutdown. ~~The surveillance frequency is modified by a Note that permits a one-time extension until October 21, 1996, for performance of the three month surveillance whose due date (with 25 percent extension) falls on September 9, 1996. This provision allows performance of the surveillance to coincide with the plant mid-cycle outage and is justified by Reference 3.~~

(continued)

Not on U2

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.12.7

Verifying, by visual inspection, that the top deck doors are in place, not obstructed, and verifying free movement of the vent assembly provides assurance that the doors are performing their function of keeping warm air out of the ice condenser during normal operation, and would not be obstructed if called upon to open in response to a DBA. The Frequency of 92 days is based on engineering judgment, which considered such factors as the following:

a. The relative inaccessibility and lack of traffic in the vicinity of the doors make it unlikely that a door would be inadvertently left open;

b. Excessive air leakage would be detected by temperature monitoring in the ice condenser; and

c. The light construction of the doors would ensure that, in the event of a DBA, air and gases passing through the ice condenser would find a flow path, even if a door were obstructed.

REFERENCES

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

2. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."

~~3. TVA Letter to NRC dated July 31, 1996 - Proposed License Amendment Containment Systems~~

Not on UL

BASES (continued)

ACTIONS

A.1

In the event shield building OPERABILITY is not maintained, shield building OPERABILITY must be restored within 24 hours. Twenty-four hours is a reasonable Completion Time considering the limited leakage design of containment and the low probability of a Design Basis Accident occurring during this time period.

B.1

The Completion Time of 8 hours is based on engineering judgment. The normal alignment for both EGTS control loops is the A-Auto position. With both EGTS control loops in A-Auto, both trains will function upon initiation of a Containment Isolation Phase A (CIA) signal. In the event of a LOCA, the annulus vacuum control system isolates and both trains of the EGTS pressure control loops will be placed in service to maintain the required negative pressure. If annulus vacuum is lost during normal operations, the A-Auto position is unaffected by the loss of vacuum. This operational configuration is acceptable because the accident dose analysis conservatively assumes the annulus is at atmospheric pressure at event initiation. (Ref. 3) A Note has been provided which makes the requirement to maintain the annulus pressure within limits not applicable during venting operations, required annulus entries, or Auxiliary Building isolations not exceeding 1 hour in duration. ~~or while Penetration 1-EQH-271-0010 or 1-EQH-271-0011 in the Shield Building dome is open until annulus pressure is restored.~~ Allowing one of the Shield Building dome penetrations to be open is based on provisions being in place to close it within fifteen minutes of LOCA initiation. Limiting the time for opening either of the penetrations to a combined total of five hours a day, six days a week keeps the amount of time the Shield Building is inoperable to approximately 60 percent of the eight hour completion time for LCO B.

~~During normal plant operation, the Annulus is maintained at a negative pressure equal to or more negative than - 5 inches water gauge (wg) by the Annulus Vacuum Control subsystem (non-safety related) of the Emergency Gas Treatment System (EGTS). One train (loop) of the Annulus Vacuum Control subsystem is operating (controls in A-Auto) and one train is in standby (controls in A-Auto Stand-by). Opening Shield Building dome Penetration 1-EQH-271-0010 or 1-EQH-271-0011 during Modes 1-4 will result in the Annulus pressure becoming more positive than the - 5 inches wg required by Technical Specification 3.6.15. When the Annulus pressure becomes more positive than - 0.812 inches wg, the EGTS control system perceives that the loop in A-Auto (i.e., the operating train) has failed. Control of Annulus pressure is then transferred to the loop in A-Auto Stand-by (i.e., the train in standby). Since the loop originally controlling Annulus pressure is perceived to have failed, only one control loop (the controller originally in A-Auto Stand-by) remains functional. If a single failure of the remaining control loop were to occur, this would result in~~

Note:
The highlighted text on this page and the following page was incorporated as part of Amendment 59. This amendment also added a series of notes to Technical Specification 3.6.15. As stated in NRC's Safety Evaluation for Amendment 59 (NRC's letter dated January 6, 2006), these controls were only applicable until WBN Unit 1 entered Mode 5 at the start of the Cycle 7 refueling outage. The highlighted text in this Bases section and the notes in Technical Specification 3.6.15 will be deleted via a future amendment to the Technical Specifications.

(continued)

Not on U2

BASES

ACTIONS

B.1 (continued)

~~both control loops failing and would render the safety-related portion of EGTS inoperable. To prevent this situation, operator action will be taken to place both EGTS control loops in the A-Auto Stand-by position when the annulus differential pressure is more positive than a - 5 inches wg. If EGTS is subsequently initiated in this configuration, both trains of EGTS will start. Absent a single failure, one EGTS control loop train will manually be returned to the A-Auto position when the Annulus differential pressure becomes more negative than - 0.812 inches wg. In addition, the remaining EGTS control loop train will be turned off, then immediately placed in the A-Auto Stand-by position (i.e., the associated isolation valves shall be closed by means of the MCR hand switch). This action is in the design and is necessary to restore the EGTS to the normal operational configuration and to prevent excess EGTS exhaust and Annulus in-leakage.~~

~~Additional assurance is administratively provided of support system operability by restricting the opening of Penetration 1-EQH-271-0010 or 1-EQH-271-0011 if in Actions for LCO 3.6.9.A *EGTS*, or 3.8.1.B *AC Sources – Operating*. If a hatch is opened and one of the above systems becomes inoperable, the hatch will be closed.~~

C.1 and C.2

If the shield building cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.15.1

Verifying that shield building annulus negative pressure is within limit (equal to or more negative than - 5 inches water gauge, value does not account for instrument error, Ref. 2) ensures that operation remains within the limit assumed in the containment analysis. The 12 hour Frequency of this SR was developed considering operating experience related to shield building annulus pressure variations and pressure instrument drift during the applicable MODES.

SR 3.6.15.2

Maintaining shield building OPERABILITY requires maintaining each door in the access opening closed, except when the access opening is being used for normal transient entry and exit. The 31 day Frequency of this SR is based on engineering judgment and is considered adequate in view of the other indications of door status that are available to the operator.

(continued)

Not on U2

BASES (continued)

LCO

Was "102"

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at ~~100.6%~~ RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2 and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

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

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressuration.

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

ACTIONS

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

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements.

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

(continued)

Not on U2

BASES

ACTIONS (continued)

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The Completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined using a conservative heat balance calculation as described above (Action A.1) and in the attachment to Reference 6. The values in Specification 3.7.1 include an allowance for instrument and channel uncertainties to the allowable RTP obtained with this algorithm.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection.

C.1 and C.2

If the Required Actions are not completed within the associated Completion Time, or if one or more steam generators have ≥ 4 inoperable MSSVs, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME OM Code, ~~Section XI~~ (Ref. 4), requires that safety and relief valve tests be

(continued)

R89

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

performed as follows: ~~in accordance with ANSI/ASME OM 1-1987 (Ref. 5). According to Reference 5, the following tests are required:~~

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria;

The **ASME OM Code** ~~ANSI/ASME Standard~~ requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. Additional test frequency requirements apply during the initial five year period. ~~as discussed in Reference 5.~~ The ASME OM Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2 correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

2. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Article NC-7000, "Overpressure Protection," Class 2 Components.

3. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency," and Section 15.4, "Condition IV - Limiting Faults."

4. American Society of Mechanical Engineers, (ASME) **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants,"** ~~Boiler and Pressure Vessel Code, Section XI.~~

~~5. ANSI/ASME OM 1-1987, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices."~~

5.~~6.~~ NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

R8q

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

The Frequency is in accordance with the Inservice Testing Program or 18 months. The 18 month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. Watts Bar FSAR, Section 10.3, "Main Steam Supply System."

2. Watts Bar FSAR, Section 6.2, "Containment Systems."

3. Watts Bar FSAR, Section 15.4.2.1, "Major Rupture of a Main Steam Line."

4. 10 CFR 100.11.

5. American Society of Mechanical Engineers, **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants,"** . ~~Boiler and Pressure Vessel Code, Section XI.~~

R89

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves

BASES

BACKGROUND

The MFRVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). The safety related function of the MFIVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. Closure of the MFIVs and associated bypass valves or MFRVs and associated bypass valves terminates flow to the steam generators. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs and associated bypass valves, or MFRVs and associated bypass valves, effectively terminates the addition of normal feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFIVs and associated bypass valves, isolate the nonsafety-related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break.

One MFIV and one MFRV are located on each 16 inch MFW line. One bypass MFRV and one bypass MFIV are located on a smaller 6 inch startup flow feedwater line. Both the MFIV and bypass MFIV are located in the main steam valve vault close to containment.

and tempering

deleted by R76

(continued)

Not on U2

BASES

ACTIONS (continued)

* See next page to see what R76 deleted.

C.1

With one MFIV or MFRV bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status within 72 hours. The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

D.1

With an MFIV and MFRV in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, at least one valve in the flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV, or otherwise isolate the affected flow path.

E.1

With two bypass valves in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, at least one valve in the flow path must be restored to OPERABLE status within 8 hours. The Completion Time of 8 hours is consistent with Condition D.

(continued)

Not on U2

BASES

ACTIONS (continued)

C.1

With one MFIV or MFRV bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status within 72 hours. ~~The inoperable valve should not be closed and isolated for long periods of time since the 6 inch bypass line provides a small tempering flow to the upper SG nozzle. This limits the temperature difference between the SG and condensate storage tank fluid which would be supplied by the AFW system. The 6 inch line may be isolated for short periods of time to support calorimetric flow measurements.~~

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

D.1

With an MFIV and MFRV in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, at least one valve in the flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV, or otherwise isolate the affected flow path.

E.1

With two bypass valves in the same flow path inoperable, there may be no redundant system to operate automatically and perform the required safety function. Under these

(continued)

BASES

ACTIONS

E.1 (continued)

conditions, at least one valve in the flow path must be restored to OPERABLE status within 8 hours. The Completion Time of 8 hours is consistent with Condition D.

F.1 and F.2

If the MFIV(s) and MFRV(s) and the associated bypass valve(s) cannot be restored to OPERABLE status, or the MFIV(s) or MFRV(s) closed, or isolated within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV, MFRV, and associated bypass valves is ≤ 6.5 seconds on an actual or simulated actuation signal. The MFIV and MFRV closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME OM Code, ~~Section XI~~ (Ref. 2), quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this SR is in accordance with the Inservice Testing Program or 18 months. The 18 month Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

(continued)

R89

BASES (continued)

REFERENCES

1. FSAR, Section 10.4.7, "Condensate and Feedwater Systems."

2. American Society of Mechanical Engineers, **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants,"** . ~~**Boiler and Pressure Vessel Code, Section XI.**~~

R89

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.2

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

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

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment. This SR is modified by a Note that states that the SR is not required in MODE 4. MODE 4 does not require automatic activation of the AFW because there is a sufficient time frame for operator action. This is based on the fact that even at 0% power (MODE 3) there is approximately a 10 minute trip delay before actuation of the AFW system to allow for operator action. In MODE 4 the heat removal requirements would be less providing more time for operator action.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow through the flow paths from the CST to each steam generator prior to entering MODE 2 after initial fuel loading and prior to subsequent entry into MODE 2 whenever the unit has been in any combination of MODES 5 or 6 for greater than 30 days. Operability of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgment and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned.

REFERENCES

1. Watts Bar FSAR, Section 10.4.9, "Auxiliary Feedwater System."

2. American Society of Mechanical Engineers, **OM Code, "Code for Operation and Maintenance of Nuclear Power Plants,"** . ~~**Boiler and Pressure Vessel Code, Section XI.**~~

R89

BASES

APPLICABLE SAFETY ANALYSES (continued)

power. Single failures that also affect this event include the following:

a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generators (requiring additional steam to drive the remaining AFW pump turbine); and

b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

A nonlimiting event considered in CST inventory determinations is a break in either the main feedwater bypass line or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

Because the CST is the preferred source of feedwater and is relied on almost exclusively for accidents and transients, the CST satisfies Criterion 3 of the NRC Policy Statement.

LCO

As the preferred water source to satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for 2 hours following a reactor trip ~~from 100.6%~~ RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

Was "102"

The CST level required is equivalent to a usable volume of ≥ 200,000 gallons, which is based on holding the unit in MODE 3 for 2 hours, followed by a cooldown to RHR entry conditions at 50°F/hour. This basis is established in Reference 4 and exceeds the volume required by the accident analysis.

(continued)

Not on UL

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS provides a protected environment from which ~~operators~~ ***occupants*** can control the unit following an uncontrolled release of radioactivity***, hazardous chemicals, or smoke***.

The CREVS consists of two independent, redundant trains that recirculate and filter the ***air in the*** control room ***envelope (CRE)*** ~~air~~ ***and a CRE boundary that limits the inleakage of unfiltered air***. Each ***CREVS*** train consists of a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, ***doors, barriers,*** and instrumentation also form part of the system.

Bases Insert 1

The CREVS is an emergency system, parts of which also operate during normal unit operations.

Actuation of the CREVS occurs automatically upon receipt of a safety injection signal in either unit or upon indication of high radiation in the outside air supply. Actuation of the system to the emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the ~~control room~~ air ***within the CRE*** through the redundant trains of air handling units, with a portion of the stream of air directed through HEPA and the charcoal filters. The emergency mode also initiates pressurization and filtered ventilation of the air supply to the ***CRE*** ~~control room~~. Pressurization of the ***CRE*** ~~control room~~ prevents infiltration of unfiltered air from the surrounding areas of the building.

A single ***CREVS*** train ***operating at a flow rate of 4000 cubic feet per minute plus or minus 10 percent (includes less than or equal to 711 cubic feet per minute pressurization flow)*** will pressurize the ***CRE*** ~~control room~~ to a minimum 0.125 inches water gauge ~~with respect to the outside atmosphere and adjacent areas~~ ***relative to external areas adjacent to the CRE boundary***. The CREVS operation in maintaining the ***CRE*** ~~control room~~ habitable is discussed in the FSAR, Section 6.4 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open

(continued)

R9

BASES

BACKGROUND (continued)

isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. A portion of the CREVS supply air ducting serving the main control room consists of round flexible ducting, triangular ducting constructed of duct board, and connecting metallic flow channels called air bars. These components are qualified to Seismic Category 1(L) requirements, which will ensure 1) the ducting will remain in place, 2) the physical configuration will be maintained such that flow will not be impeded, and 3) the ducting pressure boundary will not be lost during or subsequent to a SSE (Ref. ~~5~~***3***). The remaining portions of CREVS are designed in accordance with Seismic Category I requirements (Ref. ~~6~~***4***).

The CREVS is designed to maintain ***a habitable environment in the CRE*** ~~the control room environment~~ for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CREVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the ***CRE*** ~~control room envelope~~ ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the ***CRE occupants*** ~~control room operators~~, as demonstrated by the ***CRE*** ~~control room accident dose~~ ***occupant dose*** analyses for the most limiting design basis loss of coolant accident, fission product release presented in the FSAR, Section 15.5.3 (Ref. ~~2~~***5***).

Bases Insert 2

~~The analysis of toxic gas releases demonstrates that the toxicity limits are not exceeded in the control room following a toxic chemical release, as presented in Reference 1.~~

The worst case single active failure of a component of the CREVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREVS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two independent and redundant CREVS trains are required to be OPERABLE to ensure that at least one is available ~~assuming~~ ***if*** a single ***active*** failure disables the other train. Total system failure***, such as from a loss of both ventilation trains or from an inoperable CRE boundary,*** could result in exceeding a dose of 5 rem ***whole body or its equivalent to any part of the body to the CRE occupants*** ~~to the control room operator~~ in the event of a large radioactive release.

(continued)

R91

BASES

LCO (continued)

~~The~~ ***Each*** CREVS ***train*** is considered OPERABLE when the individual components necessary to limit ***CRE occupant*** ~~operator~~ exposure are OPERABLE~~ in both trains~~. A CREVS train is OPERABLE when the associated:

a. Fan is OPERABLE;

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

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

Bases Insert 3

~~In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.~~

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6 and during movement of irradiated fuel assemblies, ***the*** CREVS must be OPERABLE to ***ensure that the CRE will remain habitable*** ~~control operator exposure~~ during and following a DBA.

In MODE***S*** 5 ***and*** ~~or~~ 6, the CREVS is required to cope with the release from the rupture of a waste gas decay tank.

During movement of irradiated fuel assemblies, the CREVS must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1

When one CREVS train is inoperable, ***for reasons other than an inoperable CRE boundary,*** action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the ***CRE occupant*** ~~control room~~ protection function. However, the overall reliability is reduced because a ~~single~~ failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

(continued)

BASES

ACTIONS (continued)

B.1, ***B.2*** and B.~~2~~***3***

Bases Insert 4

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREVS train ***or the CRE boundary*** cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

~~C~~***D***.1 and ~~C~~***D***.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREVS train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the ***CRE*** ~~control room~~. This places the unit in a condition that minimizes ***the accident*** risk. This does not preclude the movement of fuel to a safe position.

~~D~~***E***.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, due to actions taken as a result of a tornado, the CREVS may not be capable of performing the intended function because of loss of pressurizing air to the control room. At least one train must be restored to OPERABLE status within 8 hours or the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The 8 hour restoration time is considered reasonable considering the low probability of occurrence of a design basis accident concurrent with a tornado warning.

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

(continued)

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BASES

ACTIONS

~~E~~*F*.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies with two CREVS trains inoperable ***or with one or more CREVS trains inoperable due to an inoperable CRE boundary***, action must be taken immediately to suspend activities that could result in a release of radioactivity that might ~~enter~~ ***require isolation of*** the ***CRE*** ~~control room~~. This places the unit in a condition that minimizes ***the*** accident risk. This does not preclude the movement of fuel to a safe position.

~~F~~*G*.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, for reasons other than ***Condition B or*** Condition ~~D~~***E*** the CREVS may not be capable of performing the intended function and the plant is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. The systems need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy ~~availability~~.

SR 3.7.10.2

This SR verifies that the required CREVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. ~~3~~***6***). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that each CREVS train starts and operates on an actual or simulated actuation signal. ***The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.*** ~~The Frequency of 18 months is specified in Regulatory Guide 1.52 (Ref. 3).~~

Bases Insert 5

SR 3.7.10.4

~~This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREVS. During the emergency mode of operation, the CREVS is designed to pressurize the control room ≥ 0.125 inches water gauge positive pressure with respect to the outside atmosphere and adjacent areas in order to prevent unfiltered inleakage. The CREVS is designed to maintain this positive pressure with one train at a makeup flow rate ≤ 711 cfm and a recirculation flow rate ≥ 2960 and ≤ 3618 cfm. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).~~

REFERENCES

1. Watts Bar FSAR, Section 6.4, "Habitability Systems."

Bases Insert 6

~~2. Watts Bar FSAR, Section 15.5.3, "Environmental Consequences of a Postulated Loss of Coolant Accident."~~

(continued)

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BASES

REFERENCES (continued)

~~3. Regulatory Guide 1.52, Rev. 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."~~

~~4. NUREG-0800, Standard Review Plan, Section 6.4, "Control Room Habitability System," Rev. 2, July 1981.~~

~~5. Watts Bar FSAR, Section 3.7.3.18, "Seismic Qualification of Main Control Room Suspended Ceiling and Air Delivery Components."~~

~~6. NRC Safety Evaluation dated February 12, 2004, for License Amendment 50.~~

WBN Technical Specification (TS) Change TS-07-14
Inserts for Proposed Bases Changes

Bases Insert 1:

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

Bases Insert 2:

The CREVS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 1 and 2). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 1 and 2).

Bases Insert 3:

In order for the CREVS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area.. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

Bases Insert 4:

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

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WBN Technical Specification (TS) Change TS-07-14
Inserts for Proposed Bases Changes

Bases Insert 4 (continued):

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

Bases Insert 5:

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem whole body or its equivalent to any part of the body and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3 (Ref. 7), which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 8). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 9). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

Bases Insert 6:

2. Watts Bar FSAR, Section 9.4, "Air Conditioning, Heating, Cooling, and Ventilation Systems."

3. Watts Bar FSAR, Section 3.7.3.18, "Seismic Qualification of Main Control Room Suspended Ceiling and Air Delivery Components."

4. NRC Safety Evaluation dated February 12, 2004, for License Amendment 50.

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Bases Insert 6 (continued):

5. Watts Bar FSAR, Section 15.5.3, "Environmental Consequences of a Postulated Loss of Coolant Accident."

6. Regulatory Guide 1.52, Revision 2, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."

7. Regulatory Guide 1.196, Revision 0, "Control Room Habitability at Light-Water Nuclear Power Reactors"

8. NEI 99-03, "Control Room Habitability Assessment," June 2001.

9. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694).

R91

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

BASES

BACKGROUND

The ABGTS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident and from the area of active Unit 1 ECCS components and Unit 1 penetration rooms following a loss of coolant accident (LOCA).

The ABGTS consists of two independent and redundant trains. Each train consists of a heater, a prefilter, moisture separator, a high efficiency particulate air (HEPA) filter, two activated charcoal adsorber sections for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis. The system initiates filtered ventilation of the Auxiliary Building Secondary Containment Enclosure (ABSCE) exhaust air following receipt of a Phase A containment isolation signal or a high radiation signal from the spent fuel pool area.

The ABGTS is a standby system, not used during normal plant operations. During emergency operations, the ABSCE dampers are realigned and ABGTS fans are started to begin filtration. Air is exhausted from the Unit 1 ECCS pump rooms, Unit 1 penetration rooms, and fuel handling area through the filter trains. The prefilters or moisture separators remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

INSERT 5

The ABGTS is discussed in the FSAR, Sections 6.5.1, 9.4.2, 15.0, and 6.2.3 (Refs. 1, 2, 3, and 4, respectively).

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA

(continued)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The DBA analysis of the fuel handling accident assumes that only one train of the ABGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the ABSCE is determined for a fuel handling accident and for a LOCA. These assumptions and the analysis follow the guidance provided in Regulatory Guides 1.25 (Ref. 5) and 1.4 (Ref. 6).

INSERT 6

The ABGTS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two independent and redundant trains of the ABGTS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the ABSCE exceeding the 10 CFR 100 (Ref. 7) limits in the event of a fuel handling accident or LOCA.

The ABGTS is considered OPERABLE when the individual components necessary to control exposure in the fuel handling building are OPERABLE in both trains. An ABGTS train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

c. Heater, moisture separator, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

APPLICABILITY

In MODE 1, 2, 3, or 4, the ABGTS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus.

(continued)

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BASES

APPLICABILITY (continued)

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

During movement of irradiated fuel in the fuel handling area, the ABGTS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. SEE ADDITIONAL DISCUSSION IN THE BACKGROUND AND APPLICABLE SAFETY ANALYSIS SECTIONS.

ACTIONS

A.1

With one ABGTS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the ABGTS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable ABGTS train, and the remaining ABGTS train providing the required protection.

B.1 and B.2

In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The 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.

C.1 and C.2

When Required Action A.1 cannot be completed within the required Completion Time, during movement of irradiated fuel assemblies in the fuel handling area, the OPERABLE ABGTS train must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected.

If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel

(continued)

R87

INSERT 5

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130, and -131 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. These changes will require that the containment ventilation instrumentation remain operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc are open to the Auxiliary Building ABSCE spaces. In addition, the ABGTS must remain operable if these containment penetrations are open to the Auxiliary Building during movement of irradiated fuel inside containment.

R87

INSERT 6

When moving irradiated fuel inside containment or in the Auxiliary Building with containment air locks or penetrations open to the Auxiliary Building ABSCE spaces, or when moving fuel in the Auxiliary Building with the containment equipment hatch open, the provisions to initiate a CVI from the spent fuel pool radiation monitors and to initiate an ABI (i.e., the portion of an ABI normally initiated by the spent fuel pool radiation monitors) from a CVI, including a CVI initiated by the containment purge monitors, in the event of a fuel handling accident (FHA) must be in place and functioning. The containment equipment hatch cannot be open when moving irradiated fuel inside containment in accordance with Technical Specification 3.9.4.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required. However, whether the containment purge system is operated or not in this configuration, all containment ventilation isolation valves and associated instrumentation must remain operable. This requirement is necessary to ensure a CVI can be accomplished from the spent fuel pool radiation monitors in the event of a FHA in the Auxiliary Building.

R87

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from primary to secondary leakage in the steam generator. Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

~~The secondary coolant specific activity of 0.1 µCi/gm is used as input to the steam line break accident analysis.~~ The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

Was

This limit is lower than the activity value that might be expected from a 1 gpm leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 µCi/gm (LCO 3.4.16, "RCS Specific Activity").

(continued)

Not on U2

BASES

ACTIONS (continued)

B.1 ~~and C.1~~

To ensure a highly reliable power source remains with one or more DGs inoperable in Train A OR with one or more DGs inoperable in Train B, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2 ~~and C.2~~

Required Action~~s~~ B.2 ~~and C.2 are~~ is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven auxiliary feedwater pumps. Single train systems, such as the turbine driven auxiliary feedwater pump, are not included. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has inoperable DG(s).

The Completion Time for Required Action~~s~~ B.2 ~~and C.2 are~~ is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other train (Train A or Train B) is inoperable.

If at any time during the existence of this Condition (one or more DGs inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one or more required DGs in Train A or one or more DGs in Train B inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DGs, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is Acceptable because it minimizes risk while allowing time for restoration before subjecting the plant to transients associated with shutdown.

(continued)

Not on UL

BASES

ACTIONS

B.2 ~~and C.2~~ (continued)

In this Condition, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

~~B.3.1, B.3.2, C.3.1 and C.3.2~~ B.3.1 and B.3.2

Required Action~~s~~ B.3.1 ~~and C.3.1~~ provide~~s~~ an allowance to avoid unnecessary testing of OPERABLE DG(s). If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. For the performance of a Surveillance, Required Action B.3.1 is considered satisfied since the cause of the DG being inoperable is apparent. If the cause of inoperability exists on other DG(s), the other DG(~~s~~) would be declared inoperable upon discovery and Condition F of LCO 3.8.1 would be ~~entered~~ if the other inoperable DGs are not on the same train, otherwise, if the other inoperable DGs are on the same train, the unit ~~is~~ in Condition ~~F~~. Once the failure is repaired, the common cause failure no longer exists, and Required Action~~s~~ B.3.1 ~~and B.3.2 are~~ satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG(s), performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

remains

B

is

or B.7

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1~~, B.3.2, C.3.1 or C~~.3.2, the corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 12 hour constraint imposed while in Condition B ~~or C~~.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE DGs are not affected by the same problem as the inoperable DG.

(continued)

Not on U2

According to Regulatory Guide 1.93, (Ref. 6), operation may continue in Condition B for a period that should not exceed 72 hours.

BASES

ACTIONS (continued)

B.4

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The ~~14 day~~ 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If ~~Condition~~ B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to ~~14 days~~ 72 hours. This could lead to a total of ~~17 days~~ 144 hours, since initial failure to meet the LCO, to restore the DGs. At this time, an offsite circuit could again become inoperable, the DGs restored OPERABLE, and an additional 72 hours (for a total of ~~20~~ 9 days) allowed prior to complete restoration of the LCO. The 17 day Completion Time provides a limit on time allowed in a specified ~~condition~~ after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the ~~14 day~~ 72 hour and ~~17~~ 6 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

~~Compliance with the contingency actions listed in Bases Table 3.8.1-2 is required whenever Condition B is entered for a planned or unplanned outage which will extend beyond 72 hours. If Condition B is entered initially for an activity intended to last less than 72 hours or for an unplanned outage, the contingency actions should be invoked as soon as it is established that the outage period will be longer than 72 hours. The contingency actions applicable to Surveillance Requirement (SR) 3.8.1.14 must be invoked prior to initiation of the test.~~

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

~~According to TVA's probabilistic safety analysis described in Reference 11, 12 hours is reasonable to confirm the OPERABLE DGs are not affected by the same problem as the inoperable DG.~~

(continued)

Not on U2

BASES

ACTIONS (continued)

~~C.4~~

According to Regulatory Guide 1.93, (Ref. 6), operation may continue in Condition C for a period that should not exceed 72 hours.

In Condition C, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period. Restoration of at least one DG within 72 hours results in reverting back under Condition B and continuing to track the "time zero" completion time for one DG inoperable.

The second Completion Time for Required Action C.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 144 hours, since initial failure to meet the LCO, to restore the DGs. At this time, an offsite circuit could again become inoperable, the DGs restored OPERABLE, and an additional 72 hours (for a total of 9 days) allowed prior to complete restoration of the LCO. The 6 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 6 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action C.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition C was entered.

(continued)

Not on U2

BASES

ACTIONS (continued)

D.1 and D.2

Required Action D.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. This includes motor driven auxiliary feedwater pumps. Single train features, such as the turbine driven auxiliary pump, are not included in the list.

The Completion Time for Required Action D.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition D (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable (e.g., combinations that involve an offsite circuit and one DG inoperable, or one or more DGs in each train inoperable). However, two factors tend to decrease the severity of this level of degradation:

(continued)

Not on UL

BASES

ACTIONS

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

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the plant in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

~~E~~.1 and ~~E~~.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition ~~E~~ are modified by a Note to indicate that when Condition ~~E~~ is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems - Operating," must be immediately entered. This allows Condition ~~E~~ to provide requirements for the loss of one offsite circuit and one or more DGs in a train, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition ~~E~~ for a period that should not exceed 12 hours.

(continued)

BASES

ACTIONS

~~E~~.1 and ~~E~~.2 (continued)

D

In Condition ~~E~~, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition ~~D~~ (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

C

E ~~F~~.1

With one or more required DGs in Train A inoperable simultaneous with one or more required DGs in Train B inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with one or more required DGs in Train A inoperable simultaneous with one or more required DGs in Train B inoperable, operation may continue for a period that should not exceed 2 hours.

F F

~~G~~.1 and ~~G~~.2

If the inoperable AC electric power sources cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

Not on UL

BASES

ACTIONS (continued)

~~H.1 and I.1~~ G H

Condition ~~H~~ G and Condition ~~I~~ H corresponds to a level of degradation in which all redundancy in the AC electrical power supplies cannot be guaranteed. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The plant is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3) and Regulatory Guide 1.137 (Ref. 9), as addressed in the FSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. 6800 volts is the minimum steady state output voltage and the 10 second transient value. 6800 volts is 98.6% of the nominal bus voltage of 6900 V corrected for instrument error and is the upper limit of the minimum voltage required for the DG supply breaker to close on the 6.9 kV shutdown board. The specified maximum steady state output voltage of 7260 V is 110% of the nameplate rating of the 6600 V motors. The specified 3 second transient value of 6555 V is 95% of the nominal bus voltage of 6900 V. The specified maximum transient value of 8880 V is the maximum equipment withstand value provided by the DG manufacturer. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to ± 2% of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

(continued)

Not on UL

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

~~Prior to performance of this SR in Modes 1 or 2, actions are taken to establish that adequate conditions exist for performance of the SR. The required actions are defined in Bases Table 3.8.1-2.~~

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 10 seconds. The minimum voltage and frequency stated in the SR are those necessary to ensure the DG can accept DBA loading while maintaining acceptable voltage and frequency levels. Stable operation at the nominal voltage and frequency values is also essential to establishing DG OPERABILITY, but a time constraint is not imposed. This is because a typical DG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the SR. In lieu of a time constraint in the SR, WBN will monitor and trend the actual time to reach steady state operation as a means of ensuring there is no voltage regulator or governor degradation which could cause a DG to become inoperable. The 10 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.9 (Ref. 3), Table 1.

The DG engines for WBN have an oil circulation and soakback system that operates continuously to preclude the need for a prelube and warmup when a DG is started from standby.

(continued)

Not on U2

BASES

REFERENCES (continued)

9. Regulatory Guide 1.137, Rev. 1, "Fuel Oil Systems for Standby Diesel Generators," October 1979.

10. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables.

~~11. TVA's letter to NRC dated August 7, 2001, Technical Specification Change TS-01-04, Diesel Generator (DG) Risk Informed Allowed Outage Time (AOT) Extension.~~

Not on UL

Bases Table 3.8.1-2
TS Action or Surveillance Requirement (SR) Contingency Actions

	Contingency Actions to be Implemented	Applicable TS Action or SR	Applicable Modes
1.	Verify that the offsite power system is stable. This action will establish that the offsite power system is within single-contingency limits and will remain stable upon the loss of any single component supporting the system. If a grid stability problem exists, the planned DG outage will not be scheduled.	SR 3.8.1.14 Action B.4	1, 2 1, 2, 3, 4
2.	Verify that no adverse weather conditions are expected during the outage period. The planned DG outage will be postponed if inclement weather (such as severe thunderstorms or heavy snowfall) is projected.	SR 3.8.1.14 Action B.4	1, 2 1, 2, 3, 4
3.	Do not remove from service the ventilation systems for the 6.9 kV shutdown board room, the elevation 772 transformer room, or the Unit 2 480-volt shutdown board room, concurrently with the DG, or implement appropriate compensatory measures.	Action B.4	1, 2, 3, 4
4.	Do not remove the reactor trip breakers from service concurrently during planned DG outage maintenance.	Action B.4	1, 2, 3, 4
5.	Do not remove the turbine-driven auxiliary feedwater (AFW) pump from service concurrently with a Unit 1 DG outage.	Action B.4	1, 2, 3, 4
6.	Do not remove the AFW level control valves to the steam generators from service concurrently with a Unit 1 DG outage.	Action B.4	1, 2, 3, 4
7.	Do not remove the opposite train residual heat removal (RHR) pump from service concurrently with a Unit 1 DG outage.	Action B.4	1, 2, 3, 4

Not on UL

BASES

BACKGROUND (continued)

The Primary containment exhaust is monitored by a radiation detector which provides automatic containment purge ventilation system isolation upon detecting the setpoint radioactivity in the exhaust air stream. The containment purge ventilation isolation valves will be automatically closed upon the actuation of a Containment Vent Isolation signal whenever the primary containment is being purged during normal operation or upon manual actuation from the Main Control Room (Ref. 2). Requirements for Containment Vent Isolation Instrumentation are covered by LCO 3.3.6.

APPLICABLE SAFETY ANALYSES

The Reactor Building Purge Ventilation System air cleanup units ensure that the release of radioactivity to the environment is limited by cleaning up containment exhaust during a fuel handling accident before the containment purge exhaust valves are isolated. Reactor Building Purge Ventilation System filter efficiency is one of the inputs for the analysis of the environmental consequences of a fuel handling accident. Containment isolation can only result in smaller releases of radioactivity to the environment (Ref. 1). The Containment Vent Isolation System ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment (Ref. 2). Containment Vent Isolation Instrumentation is addressed by LCO 3.3.6.

The Reactor Building Purge Air Cleanup Units satisfy Criterion 3 of the NRC Policy Statement.

INSERT 7

LCO

The safety function of the Reactor Building Purge Air Cleanup Unit is related to the initial control of offsite radiation exposures resulting from a fuel handling accident inside containment. During a fuel handling accident inside containment, the Reactor Building Purge Air Cleanup Unit provides a filtered path for cleaning up any air leaving the containment until the containment ventilation is isolated.

INSERT 8

(continued)

R87

BASES (continued)

APPLICABILITY

An initial assumption in the analysis of a fuel handling accident inside containment is that the accident occurs while irradiated fuel is being handled. Therefore, LCO 3.9.8 is applicable only at this time.

SEE ADDITIONAL DISCUSSION IN THE SAFETY ANALYSIS & LCO SECTIONS

ACTIONS

A.1 and A.2

If one Reactor Building Purge Air Cleanup Unit is inoperable, that air cleanup unit must be isolated. This places the system in the required accident configuration, thus allowing refueling to continue after verifying the remaining air cleanup unit is aligned and OPERABLE.

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

B.1

With two Reactor Building Purge Air Cleanup Units inoperable, movement of irradiated fuel assemblies within containment must be suspended. This precludes the possibility of a fuel handling accident in containment with both Reactor Building Purge Air Cleanup Units inoperable. Performance of this action shall not preclude moving a component to a safe position.

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

SURVEILLANCE REQUIREMENTS

SR 3.9.8.1

The Ventilation Filter Testing Program (VFTP) encompasses the Reactor Building Purge Air Cleanup Unit filter tests in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

(continued)

INSERT 7

In addition, during movement of irradiated fuel in the Auxiliary Building when containment is open to the Auxiliary Building spaces, a high radiation signal from the spent fuel pool accident radiation monitors will initiate a CVI.

INSERT 8

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel radiation monitors 0-RE-90-102 and -103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1-RE-90-130, and -131 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. These changes will require that the containment ventilation instrumentation remain operable when moving irradiated fuel in the Auxiliary Building if the containment air locks, penetrations, equipment hatch, etc are open to the Auxiliary Building ABSCE spaces. In addition, the ABGTS must remain operable if these containment penetrations are open to the Auxiliary Building during movement of irradiated fuel in side containment.

R87

ENCLOSURE 4

Discrepancies Identified During the Process of Marking Up the TS and TS Bases

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

1. Affected sections: TS 3.3.1 and 3.4.1; TS Bases 3.3.1 and 3.4.1

Background:

Unit 1 TS Amendment 47 amended the Unit 1 TS to allow an alternate method for the measurement of RCS total flow rate via measurement of the RCS elbow tap differential pressures. Unit 1 TS Bases Revision 60 revised the Unit 1 TS Bases to implement the changes made by Amendment 47.

The review matrix for Developmental Revision A stated the following for each section:

- TS 3.3.1: "The changes will be applied to Unit 2."
- TS Bases 3.3.1: "The changes will be applied to Unit 2."
- TS 3.4.1: "The changes will be applied to Unit 2."
- TS Bases 3.4.1: "The changes will be applied to Unit 2."

Discrepancy:

The changes should not have been incorporated in Developmental Revision A of the Unit 2 TS and TS Bases because the elbow tap differential pressure measurement method is not being used on Unit 2.

Corrective Action(s):

Developmental Revision B of the Unit 2 TS and TS Bases deleted the changes incorporated per Unit 1 TS Amendment 47 / Unit 1 TS Bases Revision 60. No further corrective action is required.

2. Affected section: TS Bases 3.6.9

Background:

Unit 1 TS Bases Revision 71 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS Amendment 59 (support steam generator replacement by allowing TEMPORARY use of penetrations in Shield Building Dome during Modes 1-4).

The review matrix for Developmental Revision A stated, "This change will NOT be applied to Unit 2."

Discrepancy:

Contrary to the above, a minor portion of Unit 1 TS Bases Revision 71 (i.e., "See TS Bases 3.6.15, Shield Building, for additional information on EGTS.") was incorporated into Developmental Revision A of the Unit 2 TS Bases.

Corrective Action:

Since this wording is nothing more than an aid to indicate where additional information on EGTS can be found, leaving this statement in the Unit 2 TS Bases has no impact on the Bases.

The wording will be left in the Unit 2 TS Bases; no further corrective action is required.

3. Affected section: TS section 3.7.1

Background:

Unit 2 TS Amendment 31 amended the Unit 1 TS to approve Power Uprate using Leading Edge Flow Meter (LEFM).

The review matrix for Developmental Revision A stated, "This change will NOT be implemented on Unit 2 at this time."

Discrepancy:

Contrary to this statement, the "58%" value in REQUIRED ACTION A.1 should have remained at "59%."

Corrective Action:

A future Developmental Revision to the Unit 2 TS will correct the "58%" value in TS LCO 3.7.1, REQUIRED ACTION A.1 to "59%."

4. Affected section: TS Bases 3.8.1

Background:

Unit 1 TS Bases Revision 50 revised the Unit 1 TS Bases to reflect changes to the Unit 1 TS approved by Unit 1 TS Amendment 39 (revise LCO 3.8.1's allowed outage time to restore an inoperable emergency diesel generator to operable status from 72 hours to 14 days).

The review matrix for Developmental Revision A stated, "This change will NOT be applied to Unit 2."

Discrepancy:

Contrary to the above, the version showing the changes incorporated for Unit 2 TS Bases 3.8.1 that was provided in Developmental Revision A of the Unit 2 TS Bases included the following verbiage at the end of SR 3.8.1.14:

> "Prior to performance of this SR in Modes 1 or 2, actions are taken to establish that adequate conditions exist for performance of the SR. The required actions are defined in Bases Table 3.8.1-2."

ENCLOSURE 4

Discrepancies Identified During the Process of Marking Up the TS and TS Bases

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

This verbiage was not shown in the markup provided for Developmental Revision A of the Unit 2 TS Bases, and should NOT have been incorporated. Table 3.8.1-2 was not added by the markup, and it was not incorporated.

Corrective Action:

A future Developmental Revision to the Unit 2 TS Bases will remove the above wording from the TS Bases for SR 3.8.1.14.

ENCLOSURE 5

Commitments

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

1. A future Developmental Revision to the Unit 2 TS will correct the "58%" value in Unit 2 TS LCO 3.7.1, REQUIRED ACTION A.1 to "59%."

2. A future Developmental Revision to the Unit 2 TS Bases will remove the "Prior to performance of this SR in Modes 1 or 2, actions are taken to establish that adequate conditions exist for performance of the SR. The required actions are defined in Bases Table 3.8.1-2." wording from the TS Bases for SR 3.8.1.14.