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April 20, 2010

10 CFR 50.4
10 CFR 50.71(e)

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

Watts Bar Nuclear Plant, Unit 1
Facility Operating License No. NPF-90
NRC Docket No. 50-390

Subject: Changes Made to the Technical Specification Bases

The purpose of this letter is to provide the NRC with copies of the changes that have been made to the Watts Bar Nuclear Plant (WBN), Unit 1, Technical Specification Bases (TS Bases), through Revision 102. This information is provided in accordance with WBN TS Section 5.6, "TS Bases Control Program," on a frequency consistent with 10 CFR 50.71(e). These changes have been implemented at WBN, Unit 1, during the period since the last update of the TS Bases on September 22, 2008, and meet the criteria described within the TS Bases Control Program for which prior NRC approval is not required. The updates to the TS Bases are provided in the enclosures listed below.

There are no regulatory commitments in this submittal. Please direct any questions concerning this matter to Kevin Casey, Senior Project Manager at (423) 751-8523.

Respectfully,

R. M. Krich

Enclosures:

1. WBN, Unit 1, Technical Specification Bases - Table of Contents
2. WBN, Unit 1, Technical Specification Bases - Changed Pages

cc (Enclosures):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant

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Enclosure 1

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LIST OF ACRONYMS

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<u>Acronym</u>	<u>Title</u>
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARO	All Rods Out
ARFS	Air Return Fan System
ADV	Atmospheric Dump Valve
BOC	Beginning of Cycle
CAOC	Constant Axial Offset Control
CCS	Component Cooling System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
CSS	Containment Spray System
CST	Condensate Storage Tank
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPD	Effective Full-Power Days
EGTS	Emergency Gas Treatment System
EOC	End of Cycle
ERCW	Essential Raw Cooling Water
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilating, and Air-Conditioning
LCO	Limiting Condition For Operation
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MSIV	Main Steam Line Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NMS	Neutron Monitoring System
ODCM	Offsite Dose Calculation Manual
PCP	Process Control Program
PIV	Pressure Isolation Valve
PORV	Power-Operated Relief Valve
PTLR	Pressure and Temperature Limits Report
QPTR	Quadrant Power Tilt Ratio
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power

LIST OF ACRONYMS

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<u>Acronym</u>	<u>Title</u>
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SR	Surveillance Requirement
UHS	Ultimate Heat Sink

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B 3.6-39	0	B 3.6-85	0
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B 3.7-27	0	B 3.7-73	0
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B 3.7-29	0	B 3.7-75	61
B 3.7-30	0	B 3.7-76	61
B 3.7-31	89	B 3.7-77	61
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B 3.7-33	89	B 3.8-2	0
B 3.7-34	0	B 3.8-3	0
B 3.7-35	41	B 3.8-4	0
B 3.7-36	0	B 3.8-5	68
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B 3.8-27	19	B 3.8-72	0
B 3.8-28	50	B 3.8-73	0
B 3.8-29	0	B 3.8-74	0
B 3.8-30	0	B 3.8-75	0
B 3.8-31	0	B 3.8-76	0
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B 3.8-50	0	B 3.8-95	0
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<u>REVISIONS</u>	<u>ISSUED</u>	<u>SUBJECT</u>
NPF-20	11-09-95	Low Power Operating License
Revision 1	12-08-95	Slave Relay Testing
NPF-90	02-07-96	Full Power Operating License
Revision 2 (Amendment 1)	12-08-95	Turbine Driven AFW Pump Suction Requirement
Revision 3	03-27-96	Remove Cold Leg Accumulator Alarm Setpoints
Revision 4 (Amendment 2)	06-13-96	Ice Bed Surveillance Frequency And Weight
Revision 5	07-03-96	Containment Airlock Door Indication
Revision 6 (Amendment 3)	09-09-96	Ice Condenser Lower Inlet Door Surveillance
Revision 7	09-28-96	Clarification of COT Frequency for COMS
Revision 8	11-21-96	Admin Control of Containment Isol. Valves
Revision 9	04-29-97	Switch Controls For Manual CI-Phase A
Revision 10 (Amendment 5)	05-27-97	Appendix-J, Option B
Revision 11 (Amendment 6)	07-28-97	Spent Fuel Pool Rerack
Revision 12	09-10-97	Heat Trace for Radiation Monitors
Revision 13 (Amendment 7)	09-11-97	Cycle 2 Core Reload
Revision 14	10-10-97	Hot Leg Recirculation Timeframe
Revision 15	02-12-98	EGTS Logic Testing
Revision 16 (Amendment 10)	06-09-98	Hydrogen Mitigation System Temporary Specification
Revision 17	07-31-98	SR Detectors (Visual/audible indication)
Revision 18 (Amendment 11)	09-09-98	Relocation of F(Q) Penalty to COLR
Revision 19 (Amendment 12)	10-19-98	Online Testing of the Diesel Batteries and Performance of the 24 Hour Diesel Endurance Run

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Revision 20 (Amendment 13)	10-26-98	Clarification of Surveillance Testing Requirements for TDAFW Pump
Revision 21	11-30-98	Clarification to Ice Condenser Door ACTIONS and door lift tests, and Ice Bed sampling and flow blockage SRs
Revision 22 (Amendment 14)	11-10-98	COMS - Four Hour Allowance to Make RHR Suction Relief Valve Operable
Revision 23	01-05-99	RHR Pump Alignment for Refueling Operations
Revision 24 (Amendment 16)	12-17-98	New action for Steam Generator ADVs due to Inoperable ACAS.
Revision 25	02-08-99	Delete Reference to PORV Testing Not Performed in Lower Modes
Revision 26 (Amendment 17)	12-30-98	Slave Relay Surveillance Frequency Extension to 18 Months
Revision 27 (Amendment 18)	01-15-99	Deletion of Power Range Neutron Flux High Negative Rate Reactor Trip Function
Revision 28	04-02-99	P2500 replacement with Integrated Computer System (ICS). Delete Reference to ERFDS as a redundant input signal.
Revision 29	03-13-00	Added notes to address instrument error in various parameters shown in the Bases. Also corrected the applicable modes for TS 3.6.5 from 3 and 4 to 2, 3 and 4.
Revision 30 (Amendment 23)	03-22-00	For SR 3.3.2.10, Table 3.3.2-1, one time relief from turbine trip response time testing. Also added Reference 14 to the Bases for LCO 3.3.2.
Revision 31 (Amendment 19)	03-07-00	Reset Power Range High Flux Reactor Trip Setpoints for Multiple Inoperable MSSVs.
Revision 32	04-13-00	Clarification to Reflect Core Reactivity and MTC Behavior.

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Revision 33	05-02-00	Clarification identifying four distribution boards primarily used for operational convenience.
Revision 34 (Amendment 24)	07-07-00	Elimination of Response Time Testing
Revision 35	08-14-00	Clarification of ABGTS Surveillance Testing
Revision 36 (Amendments 22 and 25)	08-23-00	Revision of Ice Condenser sampling and flow channel surveillance requirements
Revision 37 (Amendment 26)	09-08-00	Administrative Controls for Open Penetrations During Refueling Operations
Revision 38	09-17-00	SR 3.2.1.2 was revised to reflect the area of the core that will be flux mapped.
Revision 39 (Amendments 21 and 28)	09-13-00	Amendment 21 - Implementation of Best Estimate LOCA analysis. Amendment 28 - Revision of LCO 3.1.10, "Physics Tests Exceptions - Mode 2."
Revision 40	09-28-00	Clarifies WBN's compliance with ANSI/ANS-19.6.1 and deletes the detailed descriptions of Physics Tests.
Revision 41 (Amendment 31)	01-22-01	Power Uprate from 3411 MWt to 3459 MWt Using Leading Edge Flow Meter (LEFM)
Revision 42	03-07-01	Clarify Operability Requirements for Pressurizer PORVs
Revision 43	05-29-01	Change CVI Response Time from 5 to 6 Seconds
Revision 44 (Amendment 33)	01-31-02	Ice weight reduction from 1236 to 1110 lbs per basket and peak containment pressure revision from 11.21 to 10.46 psig.
Revision 45 (Amendment 35)	02-12-02	Relaxation of CORE ALTERATIONS Restrictions
Revision 46	02-25-02	Clarify Equivalent Isolation Requirements in LCO 3.9.4

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Revision 47 (Amendment 38)	03-01-02	RCS operational LEAKAGE and SG Alternate Repair Criteria for Axial Outside Diameter Stress Corrosion Cracking (ODSCC)
Revision 48 (Amendment 36)	03-06-02	Increase Degraded Voltage Time Delay from 6 to 10 seconds.
Revision 49 (Amendment 34)	03-08-02	Deletion of the Post-Accident Sampling System (PASS) requirements from Section 5.7.2.6 of the Technical Specifications.
Revision 50 (Amendment 39)	08-30-02	Extension of the allowed outage time (AOT) for a single diesel generator from 72 hours to 14 days.
Revision 51	11-14-02	Clarify that Shutdown Banks C and D have only One Rod Group
Revision 52 (Amendment 41)	12-20-02	RCS Specific Activity Level reduction from <1.0 μ Ci/gm to <0.265 μ Ci/gm.
Revision 53 (Amendment 42)	01-24-03	Revise SR 3.0.3 for Missed Surveillances
Revision 54 (Amendment 43)	05-01-03	Exigent TS SR 3.5.2.3 to delete SI Hot Leg Injection lines from SR until U1C5 outage.
Revision 55	05-22-03	Editorial corrections (PER 02-015499), correct peak containment pressure, and revise I-131 gap inventory for an FHA.
Revision 56	07-10-03	TS Bases for SRs 3.8.4.8 through SR 3.8.4.10 clarification of inter-tier connection resistance test.
Revision 57	08-11-03	TS Bases for B 3.5.2 Background information provides clarification when the 9 hrs for hot leg recirculation is initiated.
Revision 58 (Amendment 45)	09-26-03	The Bases for LCO 3.8.7 and 3.8.8 were revised to delete the Unit 2 Inverters.
Revision 59 (Amendment 46)	09-30-03	Address new DNB Correlation in B2.1.1 and B3.2.12 for Robust Fuel Assembly (RFA)-2.
Revision 60 (Amendment 47)	10-06-03	RCS Flow Measurement Using Elbow Tap Flow Meters (Revise Table 3.3.1-1(10) & SR 3.4.1.4).

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Revision 61 (Amendments 40 and 48)	10-14-03	Incorporated changes required to implement the Tritium Program (Amendment 40) and Stepped Boron Concentration increases for RWST and CLAs (Amendment 48) depending on the number of TPBARS installed into the reactor core.
Revision 62	10-15-03	Clarified ECCS venting in Bases Section B 3.5.2 (WBN-TS-03-19)
Revision 63	12-08-03	The contingency actions listed in Bases Table 3.8.1-2 were reworded to be consistent with the NRC Safety Evaluation that approved Tech Spec Amendment 39.
Revision 64 (Amendment 50)	03-23-04	Incorporated Amendment 50 for the seismic qualification of the Main Control Room duct work. Amendment 50 revised the Bases for LCO 3.7.10, "CREVS," and LCO 3.7.11, "CREATCS." An editorial correction was made on Page B 3.7-61.
Revision 65	04-01-04	Revised the Bases for Action B.3.1 of LCO 3.8.1 to clarify that a common cause assessment is not required when a diesel generator is made inoperable due to the performance of a surveillance.
Revision 66	05-21-04	Revised Page B 3.8-64 (Bases for LCO 3.8.4) to add a reference to SR 3.8.4.13 that was inadvertently deleted by the changes made for Amendment 12.
Revision 67 (Amendment 45)	03-05-05	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate changes to the Vital Inverters (DCN 51370). Refer to the changes made for Bases Revision 58 (Amendment 45)
Revision 68 (Amendment 55)	03-22-05	Amendment 55 modified the requirements for mode change limitations in LCO 3.0.4 and SR 3.0.4 by incorporating TSTF-359, Revision 9.

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Revision 68 (Amendment 55 and 56)	03-22-05	Change MSLB primary to secondary leakage from 1 gpm to 3 gpm (WBN-TS-03-14).
Revision 69 (Amendment 54)	04-04-05	Revised the use of the terms inter-tier and inter-rack in the Bases for SR 3.8.4.10.
Revision 70 (Amendment 58)	10-17-05	Alternate monitoring process for a failed Rod Position Indicator (RPI) (TS-03-12).
Revision 71 (Amendment 59)	02-01-06	Temporary Use of Penetrations in Shield Building Dome During Modes 1-4 (WBN-TS-04-17)
Revision 72	08-31-06	Minor Revision (Corrects Typographical Error) – Changed LCO Bases Section 3.4.6 which incorrectly referred to Surveillance Requirement 3.4.6.2 rather than correctly identifying Surveillance Requirement 3.4.6.3.
Revision 73	09-11-06	Updated the Bases for LCO 3.9.4 to clarify that penetration flow paths through containment to the outside atmosphere must be limited to less than the ABSCE breach allowance. Also administratively removed from the Bases for LCO 3.9.4 a statement on core alterations that should have been removed as part of Amendment 35.
Revision 74	09-16-06	For the LCO section of the Bases for LCO 3.9.4, administratively removed the change made by Revision 73 to the discussion of an LCO note and placed the change in another area of the LCO section.
Revision 75 (Amendment 45)	09-18-06	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate a spare inverter for Channel 1-II of the Vital Inverters (DCN 51370).

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Revision 76 (Amendment 45)	09-22-06	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate a spare inverter for Channel 1-IV of the Vital Inverters (DCN 51370).
Revision 77 (Amendment 45)	10-10-06	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate a spare inverter for Channel 1-I of the Vital Inverters (DCN 51370).
Revision 78 (Amendment 45)	10-13-06	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate a spare inverter for each of the Vital Inverters (DCN 51370).
Revision 79 (Amendment 60, 61 and 64)	11-03-06	Steam Generator Narrow Range Level Indication Increased from 6% to 32% (WBN-TS-05-06) Bases Sections 3.4.5, 3.4.6, and 3.4.7.
Revision 80	11-08-06	Revised the Bases for SR 3.5.2.8 to clarify that inspection of the containment sump strainer constitutes inspection of the trash rack and the screen functions.
Revision 81 (Amendment 62)	11-15-06	Revised the Bases for SR 3.6.11.2, 3.6.11.3, and 3.6.11.4 to address the Increase Ice Weight in Ice Condenser to Support Replacement Steam Generators (WBN-TS-05-09) [SGRP]
Revision 82 (Amendment 65)	11-17-06	Steam Generator (SG) Tube Integrity (WBN-TS-05-10) [SGRP]
Revision 83	11-20-06	Updated Surveillance Requirement (SR) 3.6.6.5 to clarify that the number of unobstructed spray nozzles is defined in the design bases.
Revision 84	11-30-06	Revised Bases 3.6.9 and 3.6.15 to show the operation of the EGTS when annulus pressure is not within limits.
Revision 85	03-22-07	Revised Bases 3.6.9 and 3.6.15 in accordance with TACF 1-07-0002-065 to clarify the operation of the EGTS.

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Revision 86	01-31-08	Figure 3.7.15-1 was deleted as part of Amendment 40. A reference to the figure in the Bases for LCO 3.9.9 was not deleted at the time Amendment 40 was incorporated into the Technical Specifications. Bases Revision 86 corrected this error (refer to PER 130944).
Revision 87	02-12-08	Implemented Bases change package TS-07-13 for DCN 52220-A. This DCN ties the ABI and CVI signals together so that either signal initiates the other signal.
Revision 88 (Amendment 67)	03-06-08	Technical Specification Amendment 67 increased the number of TPBARs from 240 to 400.
Revision 89 (Amendment 66)	05-01-08	Update of Bases to be consistent with the changes made to Section 5.7.2.11 of the Technical Specifications to reference the ASME Operation and Maintenance Code
Revision 90 (Amendment 68)	10-02-08	Issuance of amendment regarding Reactor Trip System and Engineered Safety Features Actuation System completion times, bypass test times, and surveillance test intervals
Revision 91 (Amendment 70)	11-25-2008	The Bases for TS 3.7.10, "Control Room Emergency Ventilation System (CREVS)" were revised to address control room envelope habitability.
Revision 92 (Amendment 71)	11-26-2008	The Bases for TS 3.4.15, "RCS Leakage Detection Instrumentation" were revised to remove the requirement for the atmospheric gaseous radiation monitor as one of the means for detecting a one gpm leak within one hour.
Revision 93 (Amendment 74)	02-09-2009	Updates the discussion of the Allowable Values associated with the Containment Purge Radiation Monitors in the LCO section of the Bases for LCO 3.3.6.
Revision 94 (Amendment 72)	02-23-2009	Bases Revision 94 [Technical Specification (TS)] Amendment 72 deleted the Hydrogen Recombiners (LCO 3.6.7) from the TS and moved the requirements to the Technical Requirements Manual.

TECHNICAL SPECIFICATION BASES - REVISION LISTING
 (This listing is an administrative tool maintained by WBN Licensing and may be updated
 without formally revising the Technical Specification Bases Table-of-Contents)

REVISIONS	ISSUED	SUBJECT
Revision 95	03-05-2009	Corrected an error in SR 3.3.2.6 which referenced Function 6.g of TS Table 3.3.2-1. This function was deleted from the TS by Amendment 1.
Revision 96 (Amendment 75)	06-19-2009	Modified Mode 1 and 2 applicability for Function 6.e of TS Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation." This is associated with AFW automatic start on trip of all main feedwater pumps. In addition, revised LCO 3.3.2, Condition J, to be consistent with WBN Unit 1 design bases.
Revision 97 (Amendment 76)	09-23-2009	Amendment 76 updates LCO 3.8.7, "Inverters - Operating" to reflect the installation of the Unit 2 inverters.
Revision 98 (Amendments 77, 79, & 81)	10-05-2009	Amendment 77 revised the number of TPBARs that may be loaded in the core from 400 to 704. Amendment 79 revised LCO 3.6.3 to allow verification by administrative means isolation devices that are locked, sealed, or otherwise secured. Amendment 81 revised the allowed outage time of Action B of LCO 3.5.1 from 1 hour to 24 hours.
Revision 99	10-09-2009	Bases Revision 99 incorporated Westinghouse Technical Bulletin (TB) 08-04.
Revision 100	11-17-2009	Bases Revision 100 revises the LCO description of the Containment Spray System to clarify that transfer to the containment sump is accomplished by manual actions.
Revision 101	02-09-2010	Bases Revision 101 implemented DCN 52216-A that will place both trains of the EGTS pressure control valve's hand switches in A-AUTO and will result in the valves opening upon initiation of the Containment Isolation phase A (CIA) signal. They will remain open independent of the annulus pressure and reset of the CIA.
Revision 102	03-01-2010	Bases Revision 102 implemented EDC 52564-A which addresses a new single failure scenario relative to operation of the EGTS post LOCA.

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Enclosure 2

WBN, Unit 1, Technical Specification Bases - Changed Pages

BASES (continued)

LCO

The Heat Flux Hot Channel Factor, F_q(Z), shall be limited by the following relationships:

$$F_q(Z) \leq \frac{CFQ}{P} K(Z) \quad \text{for } P > 0.5$$

$$F_q(Z) \leq \frac{CFQ}{0.5} K(Z) \quad \text{for } P \leq 0.5$$

where: CFQ is the F_q(Z) limit at RTP provided in the COLR,

K(Z) is the normalized F_q(Z) as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The actual values of CFQ and K(Z) are given in the COLR; however, CFQ is normally a number on the order of 2.4, and K(Z) is a function that looks like the one provided in Figure B 3.2.1-1.

For Relaxed Axial Offset Control operation, F_q(Z) is approximated by F_q^C(Z) and F_q^W(Z). Thus, both F_q^C(Z) and F_q^W(Z) must meet the preceding limits on F_q(Z).

An F_q^C(Z) evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value (F_q^M(Z)) of F_q(Z). Then,

$$F_q^C(Z) = F_q^M(Z) 1.0815$$

where 1.0815 is a factor that accounts for fuel manufacturing tolerances and flux map measurement uncertainty.

F_q^C(Z) is an approximation of the steady state F_q(Z).

(continued)

BASES

LCO
(continued)

The expression for F_Q^W(Z) is:

$$F_Q^W(Z) = F_Q^C(Z) W(Z)/P \text{ for } P > 0.5$$

$$F_Q^W(Z) = F_Q^C(Z) W(Z)/0.5 \text{ for } P \leq 0.5$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR.

The F_Q(Z) limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA, and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA F_Q(Z) limits. If F_Q(Z) cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for F_Q(Z) produces unacceptable consequences if a design basis event occurs while F_Q(Z) is outside its specified limits.

APPLICABILITY

The F_Q(Z) limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which F_Q^C(Z) exceeds its limit, maintains an acceptable absolute power density. F_Q^C(Z) is F_Q^M(Z) multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. F_Q^M(Z) is the measured value of F_Q(Z). The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^C(Z)$ and $F_Q^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from incore flux map results and $F_Q^C(Z) = F_Q^M(Z) 1.0815$ (Ref. 4). $F_Q^C(Z)$ is then compared to its specified limits.

The limit with which $F_Q^C(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called $K(Z)$ provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q^C(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by $\geq 10\%$ RTP since the last determination of $F_Q^C(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F_Q^C(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the F_Q(Z) limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, F_Q^C(Z), by W(Z) and dividing by P gives the maximum F_Q(Z) calculated to occur in normal operation, F_Q^W(Z). Scaling the W(Z) factors by "1/P" accounts for the possibility that reactor power may be increased prior to the next F_Q surveillance.

The limit with which F_Q^W(Z) is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. F_Q^W(Z) evaluations are not applicable for the following axial core regions, measured in percent of core height:

- a. Lower core region, from 0 to 10% inclusive; and
- b. Upper core region, from 90 to 100% inclusive.

The top and bottom 10% of the core are excluded from the evaluation because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If F_Q^W(Z) is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to F_Q^M(Z) that may occur and cause the F_Q(Z) limit to be exceeded before the next required F_Q(Z) evaluation.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.2 (continued)

If the two most recent F_Q(Z) evaluations show an increase in the expression

maximum over z

$$\left[\frac{F_Q^c(z)}{K(z)} \right],$$

it is required to meet the F_Q(Z) limit with the last F_Q^W(Z) increased by the appropriate factor specified in the COLR, or to evaluate F_Q(Z) more frequently, each 7 EFPD. These alternative requirements prevent F_Q(Z) from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the F_Q(Z) limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

F_Q(Z) is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that F_Q(Z) is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of F_Q(Z) evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
2. Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized water Reactors," May 1974.

(continued)

BASES

REFERENCES
(continued)

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."
 4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
 5. Westinghouse Technical Bulletin (TB) 08-4, "F_Q Surveillance at Part Powers," July 15, 2008.
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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.

10. Reactor Coolant Flow-Low

The Reactor Coolant Flow—Low 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-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. 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.

The LCO requires three Reactor Coolant Flow—Low channels per loop to be OPERABLE in MODE 1 above P-7.

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

(continued)

Bases

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

10. Reactor Coolant Flow-Low (continued)

in the core because of the higher power level. In MODE 1 below the P-8 setpoint and above the P-7 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since there is insufficient heat production to generate DNB conditions.

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.

(continued)

Bases

APPLICABLE
SAFETY ANALYSES
LCO, and
APPLICABILITY

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 Trip Setpoint is reached in two or more RCS loops. 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.

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 DNBR 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)

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 Trip Setpoint is reached in two or more RCS loops. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

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

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 (in two or more RCS loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

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

- Reactor Coolant Flow—Low (in two or more RCS loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

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

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 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.

(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 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 14.

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to $\leq 75\%$ RTP within 78 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 72 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 $\geq 75\%$ RTP. The 12 hour Frequency is 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. Seventy-eight hours are allowed to place the plant in MODE 3. 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. 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 12 hours while performing routine surveillance testing of other channels. The Note also allows placing

(continued)

Bases

ACTIONS

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

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 12 hour time limit is justified in 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

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 72 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 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 14.

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.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 14.

(continued)

Bases

ACTIONS
(continued)

K.1 and K.2

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to open the RTBs. Once the RTBs are open, the core is in a more stable condition and the plant enters Condition L. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to open the RTBs, are justified in Reference 7.

L.1, L.2, and L.3

Condition L applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the RTBs open. With the unit in this Condition, the NIS source range performs the monitoring and protection functions. With the required source range channel inoperable, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be closed within 1 hour as specified in LCO 3.9.2. The isolation of unborated water sources will preclude a boron dilution accident.

Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action L.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that unit conditions will change slowly.

(continued)

Bases

ACTIONS
(continued)

M.1 and M.2

Condition M applies to the following reactor trip Functions:

- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 72 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 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 Functions associated with Condition M.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 14.

N.1 and N.2

Condition N applies to the Reactor Coolant Flow—Low reactor trip Function. With one channel inoperable, the inoperable channel must be placed in trip within 72 hours. 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.

(continued)

Bases

ACTIONS

N.1 and N.2 (continued)

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.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 12 hour time limit is justified in Reference 14.

O.1 and O.2

Condition O applies to Turbine Trip on Low Fluid Oil Pressure. With one channel inoperable, the inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition 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 72 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 14.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 14.

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, 24 hours

(continued)

Bases

ACTIONS

P.1 and P.2 (continued)

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

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, 24 hours are allowed for train corrective maintenance to restore the train to OPERABLE status or the plant must be placed in MODE 3 within the next 6 hours. The 24 hour Completion Time is justified in Reference 15. 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. Placing the unit in Mode 3 results in Condition C entry while RTB(s) are inoperable.

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 coincidence logic, the associated interlock

(continued)

Bases

ACTIONS

R.1 and R.2 (continued)

must be verified to be in its required state for the existing plant condition within 1 hour or the plant must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. 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 by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

S.1 and S.2

Condition S applies to the P-7, P-8, P-9, and P-13 interlocks. With one channel inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour or the plant must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging plant systems.

T.1, T.2.1, and T.2.2

Condition T applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the plant must be placed in a MODE where the requirement does not apply. This is accomplished by placing the plant in MODE 3 within the next 6 hours (54 hours total time) followed by opening the RTBs in 1 additional hour (55 hours total time).

(continued)

Bases

ACTIONS

T.1, T.2.1, and T.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.

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 72 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 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 14.

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.

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)

Bases

ACTIONS

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

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

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 12 hour time limit is justified in Reference 14.

V.1 and V.2

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

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

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

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 12 hour time limit is justified in Reference 14.

(continued)

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

A known inoperable channel must be placed in the tripped condition within 72 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 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 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.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 12 hour time limit is justified in Reference 14.

X.1 and X.2

Condition X applies to the following reactor trip functions:

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

(continued)

Bases

ACTIONS

X.1 and X.2 (continued)

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.

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 12 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 12 hour time limit is justified in Reference 14.

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 72 hours. Since all the valves must be tripped (not fully open), in order for the reactor trip signal to be generated, it is acceptable to place more than one Turbine Stop Valve Closure channel in the trip condition. With one or more channels in the trip condition, a partial reactor trip condition exists. All of the remaining Turbine Stop Valve channels are required to actuate in order to initiate a reactor trip. If a channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced to below the P-9 setpoint within the next 4 hours. The 72 hours allowed to place an inoperable channel in the trip condition and the 4 hours allowed for reducing power are justified in Reference 14.

(continued)

Bases

ACTIONS
(continued)

Z.1

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

SURVEILLANCE
REQUIREMENTS

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

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

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

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

SR 3.3.1.1

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

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

(continued)

Bases

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.1 (continued)

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

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ³ 15% RTP and that 12 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

(continued)

Bases

**SURVEILLANCE
REQUIREMENTS**
(continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is $\geq 3\%$, 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 $f(\Delta I)$ 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 $\geq 3\%$. Note 2 clarifies that the Surveillance is required only if reactor power is $\geq 15\%$ 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.

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 62 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)

Bases

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.4 (continued)

The Frequency of every 62 days on a STAGGERED TEST BASIS is 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 92 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 92 days on a STAGGERED TEST BASIS is justified in Reference 15.

SR 3.3.1.6

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 $f(\Delta I)$ 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)

**SURVEILLANCE
REQUIREMENTS**
(continued)

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 184 days.

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

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

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

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

The Frequency of 184 days is justified in Reference 15, except for Function 13. The justification for Function 13 is provided in References 9 and 15.

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by 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)

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 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 and intermediate range channels are OPERABLE channels prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

SR 3.3.1.9

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

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

(continued)

Bases

**SURVEILLANCE
REQUIREMENTS**
(continued)

SR 3.3.1.10

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

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

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

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

SR 3.3.1.11

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

(continued)

Bases

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.11 (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.

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 exceeding the P-9 interlock whenever the unit has been in Mode 3. This Surveillance is not required if it has been performed within the previous 31 days.

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Bases

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.14 (continued)

Verification 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 exceeding the P-9 interlock.

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" (Reference 11), 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.

(continued)

Bases

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.15 (continued)

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" (Reference 12), 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.

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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.
 5. 10 CFR Part 50.49, "Environmental Qualifications of Electric 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," May 1986 and June 1990.
 8. Watts Bar Technical Requirements Manual, Section 3.3.1, "Reactor Trip System Response Times."
 9. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar, Westinghouse Letter WAT-D-10128.
 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.
 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.
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B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors: provide a measurable electronic signal or contact actuation based on the physical characteristics of the parameter being measured;
- Signal processing equipment including process protection system, and field contacts: provide analog to digital conversion (Digital Protection System), signal conditioning, setpoint comparison, process algorithm actuation (Digital Protection System), compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications; and
- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable, setpoint comparators, or contact outputs from the signal process control and protection system.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as five, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

e. Auxiliary Feedwater-Trip Of All Main Feedwater
Pumps

A Trip of both turbine driven MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no load temperature and pressure. A turbine driven MFW pump is equipped with one pressure switch on the control oil line for the speed control system. A low pressure signal from this pressure switch indicates a trip of that pump. A trip of both turbine driven MFW pumps starts the motor driven and turbine driven AFW pumps to ensure that enough water is available to act as the heat sink for the reactor.

This Function must be OPERABLE in MODES 1 and 2. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. Mode 2 applicability is when one or more turbine driven MFW pump(s) are supplying feedwater to the steam generators. In Mode 2 the AFW system pump(s) will be used for startup/shutdown conditions. During startup, a turbine driven MFW pump is placed in service along with the operating AFW System pump(s). During the process of placing the first turbine driven MFW pump in service, the anticipatory AFW auto-start channel for the non-operating turbine driven MFW pump is placed in "bypass" (electrical control circuit is de-energized) to prevent inadvertent AFW auto-start during rollup trip testing and overspeed trip testing. Once the operating turbine driven MFW pump has established sufficient feed flow to maintain SG level, the anticipatory AFW auto-start channel for the non-operating turbine driven MFW pump is placed in the "trip" condition, and the AFW pumps secured. Under these conditions, the AFW auto start circuit will be in a half trip condition (one-out-of-two) in Mode 2 and during transitions from Mode 2 to Mode 1. If the operating turbine driven MFW pump were to trip during this time period, an AFW auto start signal would be generated causing all three AFW pumps to start. Having the requirement for auto start of the AFW pumps to be required only when one or more turbine driven MFW pumps are in service limits the potential for an overcooling transient due to inadvertent AFW actuation. Mode 1 applicability allows entry into LCO 3.3.2, Condition J to be suspended for up to 4 hours when placing the second turbine driven MFW pump in service or removing one of two turbine driven MFW pumps from service. This provision will reduce administrative burden on the plant. Plant safety is not compromised during this short period because the safety grade AFW auto start channels associated with

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
LCO, and
APPLICABILITY

e. Auxiliary Feedwater-Trip Of All Main Turbine Driven Feedwater Pumps (continued)

steam generator low-low levels are operable. In MODES 3, 4, and 5, the RCPs and MFW pumps may be normally shut down, and thus neither pump trip is indicative of a condition requiring automatic AFW initiation.

f. Auxiliary Feedwater-Pump Suction Transfer on Suction Pressure-Low

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

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

These Functions must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.

7. Automatic Switchover to Containment Sump

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. The low head residual heat removal (RHR) pumps draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps.

(continued)

BASES

ACTIONS
(continued)

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

A.1

Condition A applies to all ESFAS protection functions.

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

B.1, B.2.1 and B.2.2

Condition B applies to manual initiation of:

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

This action addresses the train orientation of the SSPS for the functions listed above. If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. Note that for containment spray and Phase B

(continued)

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.

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.

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status are justified in Reference 17. 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 (30 hours total time) and in MODE 5 within an additional

(continued)

BASES

ACTIONS

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

30 hours (60 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.

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 train surveillance.

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.

If one channel is inoperable, 72 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. 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)

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 72 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.

The Required Actions have been modified by a Note that allows placing an inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 12 hours allowed for testing are justified in Reference 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

(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.

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 72 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 72 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.

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.

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

(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.

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

(continued)

BASES

ACTIONS

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

reach the 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.

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

(continued)

BASES

ACTIONS

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

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

I.1, I.2.1 and I.2.2

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

If one channel is inoperable, 72 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 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition are justified in Reference 17. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 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 reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions have been modified by a Note that allows placing an inoperable channel in bypassed condition for up to 12 hours while performing routine surveillance testing of other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. 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 turbine driven 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 an OPERABLE status. If the function cannot be returned to an

(continued)

BASES

ACTIONS

J.1 and J.2 (continued)

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.

Mode 1 applicability allows entry into LCO 3.3.2, Condition J to be suspended for up to 4 hours when placing the second turbine driven MFW pump in service or removing one of the two turbine driven MFW pumps from service.

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.

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 72 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 72 hour Completion Time is justified in References 10, 17, and 19. If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 72 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

(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 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.

(continued)

BASES

ACTIONS
(continued)

M.1.1, M.1.2 and M.2

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

A known channel inoperable, must be restored to OPERABLE status, or placed in the tripped condition within 72 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 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition are justified in Reference 17.

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.

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 12 hours while performing routine surveillance testing of other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 12-hour time limit is justified in Reference 17.

(continued)

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 12 hours while performing routine surveillance testing of the other channels. The Note also allows a channel to be placed in bypass for up to 12 hours for testing of the bypassed channel. However, only one channel may be placed in bypass at any one time. The 12 hour time limit is justified in Reference 17.

O.1 and O.2

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

If one channel is inoperable, 72 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 72 hours allowed to place the inoperable channel in the tripped condition is justified in References 10 and 17.

(continued)

BASES

ACTIONS

O.1 and O.2 (continued)

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the plant to be placed in MODE 3 within the following 6 hours. 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, these functions are no longer required OPERABLE.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 10 and 17.

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.

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.

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BASES

**SURVEILLANCE
REQUIREMENTS**
(continued)

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

SR 3.3.2.1

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

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

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

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 92 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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.2 (continued)

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 92 days on a STAGGERED TEST BASIS is 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 92 days on a STAGGERED TEST BASIS. The Frequency of 92 days is justified in Reference 18.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a COT.

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.

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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.4 (continued)

The Frequency of 184 days is justified in Reference 18, except for Function 7. The Frequency for Function 7 is justified in 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 AFW Pumps Train A and B Suction Transfer on Suction Pressure—Low (Function 6.f), and Turbine Trip and Feedwater Isolation - Main Steam Valve Vault Rooms Water Level - High (Function 5.d and 5.e).

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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.10 (continued)

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" (Reference 15), 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.

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" (Reference 16), 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.

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.

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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)

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.

BASES

LCO
(continued)

9. Containment Pressure (Wide Range)

Containment Pressure (Wide Range), a non-Type A Category 1 variable, is provided for verification of RCS and containment OPERABILITY.

Containment Pressure (Wide Range) instrumentation consists of two recorded trains on separate power supplies with a range of -5 to +60 psig.

Containment pressure wide range is used to monitor the postaccident containment pressure up to the rupture pressure of containment to indicate potential containment breach.

10. Containment Pressure (Narrow Range)

Containment Pressure (Narrow Range) is provided to determine margin to containment design pressure. The narrow range monitors are also used in event identification to monitor containment conditions following a break inside containment and to verify if the accident is being properly controlled. The narrow range instrumentation has a range of -2 to +15 psig.

11. Containment Isolation Valve Position

CIV Position, a non-Type A Category 1 variable, is provided for verification of Containment OPERABILITY, and verification of isolation after receipt of Phase A and/or Phase B isolation signals.

When used to verify valve closure for Phase A and/or Phase B isolation, the important information is the isolation status of the containment penetrations. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (i) from Table 3.3.3-1 requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve, as applicable and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

(continued)

BASES

LCO

11. Containment Isolation Valve Position (continued)

A Note to the Required Channels states that the Function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, pressure relief valve, or check valve with flow through the valve secured.

12. Containment Radiation (High Range)

Containment Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

Containment radiation level is also used to determine if a loss of reactor coolant or secondary coolant has occurred.

(continued)

BASES

LCO
(continued)

13. RCS Pressurizer Level

Pressurizer Level is one factor used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition.

Pressurizer Level instrumentation consists of the three differential pressure transmitters and associated instrumentation used to measure pressurizer level. The channels provide indication over the entire distance between taps.

14, 15. Steam Generator Water Level (Wide and Narrow Range)

SG Water Level is provided to monitor operation of decay heat removal via the SGs. The non-Type A Category 1 indication of SG level is the wide range level instrumentation.

Temperature compensation of wide range SG level indication is performed manually by the operator. The indication is cold calibrated. The uncompensated level signal is input to the plant computer for control room indications, and is used for diverse indication of AFW flow.

Narrow range steam generator level is used to make a determination on the nature of the accident in progress, e.g., verify a steam generator tube rupture. Steam generator level (Narrow Range) is also used to help identify the ruptured steam generator following a tube rupture and verify that the intact steam generators are an adequate heat sink for the reactor. Narrow range steam generator water level is used when verifying plant conditions for termination of SI during secondary plant high energy line breaks outside containment.

(continued)

BASES

LCO
(continued)

16. AFW Valve Status

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

17, 18, 19, 20. Core Exit Temperature

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

Core exit thermocouples, in conjunction with RCS wide range temperatures, are sufficient to provide indication of radial distribution of the coolant enthalpy rise across representative sections of the core. Core Exit Temperature is used to support determination of whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.

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

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

The incore thermocouple monitoring system described in Reference 4 supports the plant operating procedures.

(continued)

BASES

LCO
(continued)

21. Auxiliary Feedwater Flow

AFW Flow is provided to monitor operation of decay heat removal via the SGs.

Redundant monitoring capability is provided by two independent trains of instrumentation for each SG. Each differential pressure transmitter provides an input to a control room indicator. Since the primary indication used by the operator during an accident is the control room indicator, the PAM specification deals specifically with this portion of the instrument channel.

AFW flow is used three ways:

- to verify AFW flow to the SGs;
- to determine whether to terminate SI if still in progress, in conjunction with SG water level (narrow range); and
- to regulate AFW flow so that the SG tubes remain covered.

22. Reactor Coolant System Subcooling Margin Monitor

The RCS subcooling margin monitor is used to determine the temperature margin to saturation of the primary coolant. Control room indications are provided through the ICCM plasma display and digital panel meters. The ICCM plasma display is the primary indication used by the operator during an accident.

(continued)

BASES

LCO
(continued)

23. Refueling Water Storage Tank Level

RWST water level is used to verify the water source availability to the ECCS and Containment Spray (CS) Systems. It alerts the operator to manually switch the CS suction from the RWST to the containment sump. It may also provide an indication of time for initiating cold leg recirculation from the sump following a LOCA.

24. Steam Generator Pressure

Steam pressure is used to determine if a high energy secondary line rupture has occurred and the availability of the steam generators as a heat sink. It is also used to verify that a faulted steam generator is isolated. Steam pressure may be used to ensure proper cooldown rates or to provide a diverse indication for natural circulation cooldown.

25. Auxiliary Building Passive Sump Level

Auxiliary Building Passive Sump Level, a non-Type A Category 1 variable, monitors the sump level in the auxiliary building. The two functions of this indication are to monitor for a major breach of the spent fuel pit and to monitor for an RCS breach in the auxiliary building (i.e., an RHR or CVCS line break). The purpose is to verify that radioactive water does not leak to the auxiliary building. The Auxiliary Building Passive Sump Level monitor consists of two channels on separate power supply. One channel is recorded. The calibrated range of the two monitors are 12.5" to 72.5".

(continued)

BASES (continued)

APPLICABILITY The PAM instrumentation LCO is applicable as shown in Table 3.3.3-1. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies when one or more Functions have one required channel that is inoperable. Required Action A.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

Condition A is modified by a Note that excludes single channel Functions 3, 4, 14, and 16.

(continued)

BASES

ACTIONS
(continued)

B.1

Condition B applies when the Required Action and associated Completion Time for Condition A are not met. This Required Action specifies initiation of actions in Specification 5.9.8, "PAMS Report," which require a written report to be submitted to the NRC immediately. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

C.1

Condition C applies when one or more Functions have two inoperable required channels (i.e., two channels inoperable in the same Function). Condition C also applies to single channel Functions 3, 4, 14, and 16 when the one required channel is inoperable. Required Action C.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days.

The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function or the single required channel inoperable in the single channel Functions is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

(continued)

BASES

ACTIONS
(continued)

D.1

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

E.1 and E.2

If the Required Action and associated Completion Time of Conditions C are not met and Table 3.3.3-1 directs entry into Condition E, the plant must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and 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.

(continued)

BASES

ACTIONS
(continued)

F.1

Alternate means may be temporarily installed for monitoring reactor vessel water level and Containment Area Radiation if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. Alternate means would be developed and tested prior to use. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.9.8, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure 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. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, 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. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.3.1 (continued)

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

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 measured parameter with the necessary range and accuracy. This SR is modified by two Notes. Note 1 excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." Note 2 indicates that Function 11 and 16 (valve position indicators) are excluded from the CHANNEL CALIBRATION. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.3.3

SR 3.3.3.3 is the performance of a TADOT. This test is performed every 18 months. The test checks operation of the containment isolation valve position indicators and AFW valve position indicators. The Frequency is based on the known reliability of the indicators and has been shown to be acceptable through operating experience.

This SR has been modified by two Notes. Note 1 excludes verification of setpoints for the valve position indicators. Note 2 indicates that this SR is only applicable to Functions 11 and 16, which are the only Functions with valve position indicators.

REFERENCES

1. NUREG-0847, Safety Evaluation Report, Supplement Number 9, June 16, 1992, Section 7.5.2, "Post Accident Monitoring System."
 2. Regulatory Guide 1.97, Revision 2, December 1980, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."
 3. NUREG-0737, "Clarification of TMI Action Plan Requirements," Supplement 1, January 1983.
 4. Submittal from John H. Garrity to U.S. Nuclear Regulatory Commission dated January 24, 1992, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - NUREG 0737, Item II.F.2 - Instrumentation for Detection of Inadequate Core Cooling (ICC) - Proposed License Condition 3 (TAC Numbers M77132 and M77133).
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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.

Table 3.3.6-1 specifies the two Allowable Values (AVs) for the Containment Purge Exhaust Radiation Monitors. One AV is applicable in MODES 1, 2, 3, and 4 and the second AV is applicable during the movement of irradiated fuel assemblies inside containment when the potential for a fuel handling accident (FHA) exists. Both of these AVs are based on expected concentrations for a small break LOCA, which is more restrictive than the 10 CFR 100 limits. In addition, the source term for an FHA is significantly greater than the source term for a small break LOCA which would result in the containment purge monitors responding more quickly to the higher FHA source term (Ref. 5). Both of the specified AVs are 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 AVs. If the setpoint does not exceed the applicable AV, 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)

BASES

APPLICABILITY
(continued)

While in MODES 5 and 6 without fuel handling in progress, the Containment the Containment Vent Isolation Instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

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 3.3.6-1. 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.

A.1

Condition A applies to the failure of one containment purge isolation radiation monitor channel. Since the two containment radiation monitors are both gaseous detectors, failure of a single channel may result in loss of the redundancy. Consequently, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

(continued)

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.

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

SR 3.3.6.2

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 92 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference 4.

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 92 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference 4.

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

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.6.4

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment vent system isolation. The setpoint shall be left consistent with the current unit specific calibration procedure tolerance.

SR 3.3.6.5

SR 3.3.6.5 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 acceptable based on instrument reliability and industry operating experience.

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 3. 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 3.

(continued)

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.
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.
5. Calculation WBNTSR-038, "Required Response Time, Range and Accuracy for the Containment Purge Exhaust Radiation Monitors."

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

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

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment pocket sump used to collect unidentified LEAKAGE is instrumented to alarm for increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivity of 10^{-9} $\mu\text{Ci/cc}$ radioactivity for particulate monitoring is practical for this leakage detection system. A radioactivity detection system is included for monitoring particulate activity because of its sensitivity and rapid response to RCS LEAKAGE.

An atmospheric gaseous radioactivity monitor will provide a positive indication of leakage in the event that high levels of reactor coolant gaseous activity exist due to fuel cladding defects. The effectiveness of the atmospheric gaseous radioactivity monitors depends primarily on the activity of the reactor coolant and also, in part, on the containment volume and the background activity level. Shortly after startup and also during steady state operation with low levels of fuel defects, the level of radioactivity in the reactor coolant may be too low for the containment atmosphere gaseous radiation monitors to detect a reactor coolant leak of 1 gpm within one hour. Atmospheric gaseous radioactivity monitors are not required by this LCO.

The sample lines supplying the radioactivity monitoring instrumentation are heated (heat traced) to ensure that a representative sample can be obtained. During periods when the heat tracing is inoperable, the particulate channel of the radioactivity monitoring instrumentation is inoperable and grab samples for particulates may not be taken using the sample lines.

(continued)

BASES

BACKGROUND
(continued)

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment pocket sump. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE
SAFETY ANALYSES

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3).

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak detrimental to the safety of the unit and the public occur. RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

One method of protecting against large RCS leakage derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment pocket sump level monitor, in combination with a particulate radioactivity monitor provides an acceptable minimum.

The sample lines supplying the radioactivity monitoring instrumentation are heated (heat traced) to ensure that a representative sample can be obtained.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is to be $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With the required containment pocket sump level monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere particulate radioactivity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

Restoration of the required containment pocket sump level monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

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

With the particulate containment atmosphere radioactivity monitoring instrumentation channel inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

During periods when the heat tracing is inoperable for the sample lines supplying the radioactivity monitoring instrumentation, the particulate channel of the instrumentation is inoperable and grab samples for particulates may not be taken using the sample lines.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere particulate radioactivity monitor.

(continued)

BASES

ACTIONS

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

The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

If a Required Action of Condition A or B cannot be met, 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 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.

D.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE
REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere particulate radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

(continued)

BASES

**SURVEILLANCE
REQUIREMENTS**
(continued)

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere particulate radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and the relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."
2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Revision 0, May 1973.
3. Watts Bar FSAR, Section 5.2.7, "RCPB Leakage Detection Systems."

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The accumulators satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures \leq 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F.

In MODE 3, with RCS pressure \leq 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

(continued)

BASES (continued)

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break for the majority of plants. Even if they do discharge, their impact is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 24 hours. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 24 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 24 hours allowed to restore an inoperable accumulator to OPERABLE status is justified in WCAP-15049-A, Rev. 1 (Ref. 6).

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated 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 MODE 3 within 6 hours and pressurizer pressure reduced to ≤ 1000 psig within 12 hours.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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.

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

SR 3.5.1.1

Each accumulator valve should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator (refer to the note below). This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

Note:

In the discussion contained in the Applicable Safety Analyses of this Bases section, the borated water volume and nitrogen cover pressure specified for SR 3.5.1.2 and SR 3.5.1.3 account for instrument accuracy (Ref. 7).

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after a 75 gallons (1% volume) increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is ≥ 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA. This design feature still exists, but is no longer required for accident mitigation.

BASES

- REFERENCES
1. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."
 2. Watts Bar FSAR, Section 6.3, "Emergency Core Cooling System."
 3. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Plants."
 4. Watts Bar FSAR, Section 15.0, "Accident Analysis."
 5. NUREG-1366, Improvements to Technical Specifications Surveillance Requirements, December 1992.
 6. WCAP-15049-A, Rev. 1, April 1999
 7. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

- a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;
- b. Rod ejection accident;
- c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and
- d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. Approximately 3.0 hours after event initiation, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

(continued)

BASES

BACKGROUND
(continued)

When the suction for the ECCS and Containment Spray System pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

- a. The RWST contains sufficient borated water to support the ECCS during the injection phase;
- b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and
- c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE
SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS-Operating;" B 3.5.3, "ECCS-Shutdown;" and B 3.6.6, "Containment Spray Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available

(continued)

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 absorber rods (TPBARs) installed in the reactor core. Watts Bar is authorized to place a maximum of 704 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)

BASES

LCO
(continued)

times in the FSAR (Ref. 2). The normally closed containment isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 2.

Purge valves with resilient seals and shield building bypass valves meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator (licensed or unlicensed) at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. For valve controls located in the control room, an operator (other than the Shift Operations Supervisor (SOS), ASOS, or the Operator at the Controls) may monitor containment isolation signal status rather than be stationed at the valve controls. Other secondary responsibilities which do not prevent adequate monitoring of containment isolation signal status may be performed by the operator provided his/her primary responsibility is rapid isolation of the penetration when needed for containment isolation. Use of the Unit Control Room Operator (CRO) to perform this function should be limited to those situations where no other operator is available.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate

(continued)

BASES

ACTIONS
(continued)

compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event the isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable except for purge valve or shield building bypass leakage not within limit, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment

(continued)

BASES

ACTIONS

B.1 (continued)

and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low. Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

specifically address those penetration flow paths in a closed system. Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

D.1

With the shield building bypass leakage rate not within limit, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit within 4 hours. Restoration can be accomplished by isolating the penetration(s) that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of shield building bypass leakage to the overall containment function.

E.1, E.2, and E.3

In the event one or more containment purge valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits, or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve, or blind flange. A purge valve with resilient seals utilized to satisfy Required Action E.1 must have been demonstrated to meet the leakage

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

requirements of SR 3.6.3.5. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action E.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification that those isolation devices outside containment potentially capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the containment purge valve with resilient seal that is isolated in accordance with Required Action E.1, SR 3.6.3.5 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.5, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 3). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

Required Action E.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

F.1 and F.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

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 3.6.3.1

This SR ensures that the purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. All purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside containment, the containment annulus, and the Main Steam Valve Vault Rooms, and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves in areas where the valves are capable of being mispositioned are in the correct position. Since verification of valve position for these valves is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.2 (continued)

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located inside containment, the containment annulus, and the Main Steam Valve Vault Rooms, and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For these containment isolation valves, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls (eg: locked valve program) and may be verified by administrative means, because the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each power operated and automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The

(continued)

BASES

BACKGROUND
(continued)

The operation of the ice condenser, is adequate to assure pressure suppression during the initial blowdown of steam and water from a DBA. During the post blowdown period, the Air Return System (ARS) is automatically started. The ARS returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice and by the Containment Spray System after the ice has melted.

The Containment Spray System limits the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment.

APPLICABLE
SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2).

The DBA analyses show that the maximum peak containment pressure of 11.01 psig results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis. The calculated transient containment atmosphere temperatures are acceptable for the DBA SLB.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure signal setpoint to achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time of 221 seconds is composed of signal delay, diesel generator startup, and system startup time.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 3).

Inadvertent actuation of the Containment Spray System is evaluated in the analysis, and the resultant reduction in containment pressure is calculated. The maximum calculated steady state pressure differential relative to the Shield Building annulus is 1.4 psid, which is below the containment design external pressure load of 2.0 psid.

The Containment Spray System satisfies Criterion 3 of the NRC Policy Statement.

LCO

During a DBA, one train of Containment Spray System and RHR Spray System is required to provide the heat removal capability assumed in the safety analyses. To ensure that these requirements are met, two containment spray trains and two RHR spray trains must be OPERABLE with power from two safety related, independent power supplies. Therefore, in the event of an accident, at least one train in each system operates.

Each containment spray train typically includes a spray pump, header, valves, a heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and transferring suction to the containment sump. This suction path realignment is accomplished by manual operator action upon receipt of a Low-Low level alarm for the RWST.

(continued)

BASES

LCO
(continued)

Each RHR spray train includes a pump, header, valves, a heat exchanger, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the containment sump and supplying flow to the spray header.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the Containment Spray System. A Note has been added which states the RHR spray trains are not required in MODE 4. The containment spray system does not require supplemental cooling from the RHR spray in MODE 4.

In MODES 5 and 6, the probability and consequences of these events are reduced because of the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1 and B.1

With one containment spray train and/or RHR spray train inoperable, the affected train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

C.1 and C.2

If the affected containment spray train and/or RHR spray train 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 84 hours. The allowed Completion Times are reasonable, based

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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 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 the American Society of Mechanical Engineers (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.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 *The Bases for Specification 3.6.7 have been Deleted*

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B 3.6-44

Revision 61, 94
Amendment 40, 72

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Watts Bar-Unit 1

B 3.6-46

Revision 68, 94
Amendment 55, 72

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Watts Bar-Unit 1

B 3.6-47

Revision 94
Amendment 72

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.9 Emergency Gas Treatment System (EGTS)

BASES

BACKGROUND

The EGTS is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1), to ensure that radioactive materials that leak from the primary containment into the shield building (secondary containment) following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The containment has a secondary containment called the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a loss of coolant accident (LOCA). This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The EGTS establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. Shield building OPERABILITY is required to ensure retention of primary containment leakage and proper operation of the EGTS.

The EGTS consists of two separate and redundant trains. Each train includes a heater, a prefilter, moisture separators, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. The moisture separators function to reduce the moisture content of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a safety injection (SI) signal. The system is described in Reference 2.

(continued)

BACKGROUND
(continued)

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

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

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

APPLICABLE
SAFETY ANALYSES

The EGTS design basis is established by the consequences of the limiting DBA, which is a LOCA. The accident analysis (Ref. 3) considers two different single failure scenarios. The first one assumes that only one train of the EGTS is functional due to a postulated single failure that disables the other train. An alternate scenario assumes a single failure of the pressure control loop associated with one train of PCOs. The first scenario is bounding for thyroid dose while the alternate scenario is bounding for beta and gamma doses. The accident analysis accounts for the reduction in airborne radioactive material provided by the number of filter trains in operation for each failure scenario. The amount of fission products available for release from containment is determined for a LOCA.

The safety analysis conservatively assumes the annulus is at atmospheric pressure prior to the LOCA. The analysis further assumes that upon receipt of a Containment Isolation Phase A (CIA) signal from the RPS, the EGTS fans automatically start and achieve a minimum flow of 3600 cfm per train within 18 seconds (20 seconds from the initiating event.) This does not include 10 seconds for diesel generator startup. The analysis shows that the annulus pressure will rise to a positive value and then decrease to the EGTS control point for a single failure of one EGTS train, or slightly more negative for a single failure of a pressure control loop associated with one train of PCOs. 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 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. 6)

The EGTS satisfies Criterion 3 of the NRC Policy Statement.

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.6.9.3

The automatic startup ensures that each EGTS train responds properly. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the EGTS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.9.1.

SR 3.6.9.4

The proper functioning of the fans, dampers, filters, adsorbers, etc., as a system is verified by the ability of each train to produce the required system flow rate within the specified timeframe. The 18 month Frequency on a STAGGERED TEST BASIS is consistent with Regulatory Guide 1.52 (Ref. 4) guidance for functional testing.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."
 2. Watts Bar FSAR, Section 6.5, "Fission Product Removal and Control Systems."
 3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
 4. Regulatory Guide 1.52, Rev. 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Absorption Units of Light-Water Cooled Nuclear Power Plants."
 5. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."
 6. DCN 52216-A, "Elimination of A-AUTO STANDBY Hand Switch Position for EGTS Pressure Control Loops."
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Air Return System (ARS)

BASES

BACKGROUND

The ARS is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a Design Basis Accident (DBA). The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting post accident pressure and temperature in containment to less than design values. Limiting pressure and temperature reduces the release of fission product radioactivity from containment to the environment in the event of a DBA.

The ARS provides post accident hydrogen mixing in selected areas of containment. The ARS draws air from the dome of the containment vessel, from the reactor cavity, and from the ten dead ended (pocketed) spaces in the containment where there is potential for the accumulation of hydrogen. The minimum design flow from each potential hydrogen pocket is sufficient to limit the local concentration of hydrogen.

The ARS consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, associated damper, and hydrogen collection headers. Each train is powered from a separate Engineered Safety Features (ESF) bus.

The ARS fans are automatically started by the containment isolation Phase B signal 8 to 10 minutes after the containment pressure reaches the pressure setpoint. The time delay ensures that no energy released during the initial phase of a DBA will bypass the ice bed through the ARS fans into the upper containment compartment.

After starting, the fans displace air from the upper compartment to the lower compartment, thereby returning the air that was displaced by the high energy line break blowdown from the lower compartment and equalizing pressures throughout containment. After discharge into the lower compartment, air flows with steam produced by residual heat

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.15 Shield Building

BASES

BACKGROUND The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects containment leakage that may occur following a loss of coolant accident (LOCA) as well as other design basis accidents (DBAs) that release radioactive material. This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The Emergency Gas Treatment System (EGTS) establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment. The shield building is required to be OPERABLE to ensure retention of containment leakage and proper operation of the EGTS.

APPLICABLE SAFETY ANALYSES The design basis for shield building OPERABILITY is a LOCA. Maintaining shield building OPERABILITY ensures that the release of radioactive material from the containment atmosphere is restricted to those leakage paths and associated leakage rates assumed in the accident analyses.

The shield building satisfies Criterion 3 of the NRC Policy Statement.

LCO Shield building OPERABILITY must be maintained to ensure proper operation of the EGTS and to limit radioactive leakage from the containment to those paths and leakage rates assumed in the accident analyses.

APPLICABILITY Maintaining shield building OPERABILITY prevents leakage of radioactive material from the shield building. Radioactive material may enter the shield building from the containment following a DBA. Therefore, shield building OPERABILITY is required in MODES 1, 2, 3, and 4 when DBAs could release radioactive material to the containment atmosphere.

In MODES 5 and 6, the probability and consequences of these events are low due to the Reactor Coolant System temperature and pressure limitations in these MODES. Therefore, shield building OPERABILITY is not required in MODE 5 or 6.

(continued)

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)

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)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.15.3

This SR would give advance indication of gross deterioration of the concrete structural integrity of the shield building. The Frequency of this SR is the same as that of SR 3.6.1.1. The verification is done during shutdown.

SR 3.6.15.4

The EGTS is required to maintain a pressure equal to or more negative than - 0.50 inches of water gauge ("wg) in the annulus at an elevation equivalent to the top of the Auxiliary Building. At elevations higher than the Auxiliary Building, the EGTS is required to maintain a pressure equal to or more negative than - 0.25 "wg. The low pressure sense line for the pressure controller is located in the annulus at elevation 783. By verifying that the annulus pressure is equal to or more negative than - 0.61 "wg at elevation 783, the annulus pressurization requirements stated above are met. The ability of a EGTS train with final flow ≥ 3600 and ≤ 4400 cfm to produce the required negative pressure during the test operation provides assurance that the building is adequately sealed. The negative pressure prevents leakage from the building, since outside air will be drawn in by the low pressure at a maximum rate ≤ 250 cfm. The 18 month Frequency on a STAGGERED TEST BASIS is consistent with Regulatory Guide 1.52 (Ref. 1) guidance for functional testing.

REFERENCES

1. Regulatory Guide 1.52, Revision 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."
 2. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."
 3. DCN 52216-A, "Elimination of A-AUTO STANDBY Hand Switch Position for EGTS Pressure Control Loops."
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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 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) 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.

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.

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 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. Pressurization of the CRE prevents infiltration of unfiltered air from the surrounding areas of the building.

(continued)

BASES

BACKGROUND
(continued)

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 to a minimum 0.125 inches water gauge relative to external areas adjacent to the CRE boundary. The CREVS operation in maintaining the CRE 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 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. 3). The remaining portions of CREVS are designed in accordance with Seismic Category I requirements (Ref. 4).

The CREVS is designed to maintain a habitable environment in the CRE 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 ensures an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the CRE occupants, as demonstrated by the CRE 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. 5).

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).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

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 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 in the event of a large radioactive release.

Each CREVS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. 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.

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.

(continued)

BASES (continued)

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 during and following a DBA.

In MODES 5 and 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 protection function. However, the overall reliability is reduced because a 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.

B.1, B.2 and B.3

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.

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

(continued)

BASES (continued)

ACTIONS
(continued)

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.

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.

D.1 and 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 D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

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.

(continued)

BASES

ACTIONS

E.1 (continued)

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.

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 require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

G.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4, for reasons other than Condition B or Condition 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.

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. 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.

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air leakage 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 leakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air leakage 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 leakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

REFERENCES

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

(continued)

BASES

REFERENCES
(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).
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B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

BASES

BACKGROUND The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of two independent and redundant trains that provide cooling of recirculated control room air. Each train consists of an air handling unit (AHU), water chiller, chilled water pump, and associated piping, ductwork, instrumentation, and controls to provide for control room temperature control. The CREATCS is a subsystem providing air temperature control for the control room.

The CREATCS is an emergency system, parts of which also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room between 60°F and 104°F. The CREATCS operation in maintaining the control room temperature is discussed in the FSAR, Section 9.4.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES The design basis of the CREATCS is to maintain the control room temperature for 30 days of continuous occupancy.

The CREATCS components are arranged in redundant, safety related trains. During emergency operation, the CREATCS maintains the temperature between 60°F and 104°F. A single active failure of a component of the CREATCS, with a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. A portion of the CREATCS 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. 2). The remaining portions of CREATCS are designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. There are two unit inverters (Unit 1 and Unit 2) and one spare inverter per channel, each of which is capable of supplying its associated AC vital buses, making a total of twelve inverters. The function of the inverter is to provide AC electrical power to the AC vital buses. The inverters can be powered from an internal AC source/rectifier or from the vital battery. The vital battery provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). The spare inverters will be used as spare uninterruptible power sources; however they will not have a regulated transformer bypass source. Specific details on inverters and their operating characteristics are found in the Watts Bar FSAR, Section 8 (Ref. 1).

APPLICABLE SAFETY ANALYSES The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Section 6 (Ref. 2) and Section 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the plant. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and
- b. A worst case single failure.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

Inverters are a part of the distribution systems and, as such, satisfy Criterion 3 of the NRC Policy Statement.

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (A00) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The twelve inverters (one Unit 1, one Unit 2 and one spare per channel) ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 6.9 kV shutdown boards are de-energized.

OPERABLE inverters require the associated AC vital bus to be powered by an inverter with output voltage and frequency within tolerances and power input to the inverter from a 125 VDC vital battery. Alternatively, power supply may be from an internal AC source via rectifier as long as the vital battery is available as the uninterruptible power supply. The unit inverters have an associated bypass supply provided by a regulated transformer that is automatically connected to the associated AC vital bus in the event of inverter failure or overload. The bypass supply is not battery-backed and thus does not meet requirements for inverter operability. The spare inverters do not have an associated bypass supply. Additionally, the inverter channel must not be connected to the cross train 480 V power supply.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

(continued)

BASES (continued)

ACTIONS

A.1

With one inverter in a channel inoperable, its associated AC vital bus becomes inoperable until it is re-energized from its associated regulated transformer bypass source, inverter internal AC source, or spare inverters.

For this reason, a Note has been included in Condition A requiring the entry into the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems-Operating." This ensures that the vital bus is reenergized within 2 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the plant is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its associated regulated transformer bypass source it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices. Alternatively, an inverter may be restored to OPERABLE status by substituting its spare inverters and performing the required surveillance.

B.1 and B.2

If the inoperable devices or components 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)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed including those from the associated vital battery boards and 480 V shutdown boards, and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. Upon placing a spare inverter in service, the spare inverter is considered inoperable until this surveillance is completed. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. Watts Bar FSAR, Section 8.3.1, "AC Power System."
 2. Watts Bar FSAR, Section 15, "Accident Analysis," and Section 6 "Engineered Safety Features."
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Section 6 (Ref. 1) and Section 15 (Ref. 1), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each AC vital bus during MODES 5 and 6 ensures that:

- a. The plant can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the plant status; and
- c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO The inverters ensure the availability of electrical power for the instrumentation for systems required to shutdown the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 6.9 kV shutdown boards are de-energized. OPERABILITY of the inverters requires that the AC vital buses required by LCO 3.8.10, "Distribution Systems - Shutdown" be powered by the inverter. As a minimum, either the channel I and III or II and IV inverters for each unit (or spare inverters) shall be OPERABLE to support the distribution systems required by LCO 3.8.10. The unit inverters have an associated bypass supply provided by a regulated transformer that is automatically connected to the associated AC vital bus in the event of inverter failure or overload. The bypass supply is not battery-backed and thus does not meet requirements for inverter operability. The spare inverters do not have an associated bypass supply. Additionally, the inverter channel must not be connected to the cross-train 480 V power supply. This ensures the availability of sufficient inverter power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

- a. Systems needed to mitigate a fuel handling accident are available;
- b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, the remaining OPERABLE Inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate

(continued)
