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

November 3, 2014

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: **WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 – PERIODIC SUBMISSION FOR CHANGES MADE TO THE WBN TECHNICAL SPECIFICATION BASES AND TECHNICAL REQUIREMENTS MANUAL**

References: Tennessee Valley Authority (TVA) letter to the Nuclear Regulatory Commission (NRC) "Changes Made to the Technical Specifications Bases and Technical Requirements Manual" dated April 29, 2013

The purpose of this letter is to provide the Nuclear Regulatory Commission (NRC) with copies of changes to the WBN Technical Specification (TS) Bases, through Revision 121, and WBN Technical Requirements Manual (TRM), through Revision 54, in accordance with WBN TS Section 5.6, "TS Bases Control Program," and WBN TRM Section 5.1, "Technical Requirements Control Program," respectively. These changes have been implemented at WBN during the period since WBN's last update (April 29, 2013) and meet the criteria described within the above control programs for which prior NRC approval is not required. Both control programs require such changes to be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Per the provisions of 10 CFR 50.71(e), the Updated Final Safety Analysis Report will be provided in a separate letter. The WBN TS Bases and TRM updates for the table of contents and change pages are provided in the enclosures.

There are no new regulatory commitments in this letter. If you have questions regarding this letter, please call Gordon Arent, Director of Watts Bar Site Licensing, at (423) 365-2004.

I certify the information provided accurately presents changes made since the last TS Bases and TRM update was submitted on April 29, 2013.

Respectfully,

Kevin T. Walsh
Site Vice President
Watts Bar Nuclear Plant

Enclosures:

1 - WBN Technical Specification Bases - Table of Contents
2 - WBN Technical Specifications Bases - Changed Pages
3 - WBN Technical Requirements Manual - Table of Contents
4 - WBN Technical Requirements Manual - Changed Pages

cc (Enclosures):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant, Unit 1
NRC Senior Resident Inspector - Watts Bar Nuclear Plant, Unit 2

cc (Enclosures):

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ENCLOSURE 1
WBN TECHNICAL SPECIFICATION BASES
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Acronym	Title
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
PDMS	Power Distribution Monitoring System
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

Acronym	Title
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 3.8-37	0	B 3.8-82	97
B 3.8-38	0	B 3.8-83	97
B 3.8-39	0	B 3.8-84	75
B 3.8-40	0	B 3.8-85	0
B 3.8-41	0	B 3.8-86	97
B 3.8-42	0	B 3.8-87	75
B 3.8-43	0	B 3.8-88	0
B 3.8-44	0	B 3.8-89	78
B 3.8-45	0	B 3.8-90	0
B 3.8-46	0	B 3.8-91	78
B 3.8-47	55	B 3.8-92	0
B 3.8-48	55	B 3.8-93	78
B 3.8-49	0	B 3.8-94	0
B 3.8-50	0	B 3.8-95	0
B 3.8-51	29	B 3.8-96	0
B 3.8-52	106	B 3.8-97	0
B 3.8-53	29	B 3.8-98	33
B 3.8-54	105	B 3.8-99	0
B 3.8-55	0	B 3.8-100	0
B 3.8-56	113	B 3.8-101	0
B 3.8-57	113	B 3.8-102	0
B 3.8-58	0	B 3.9-1	0
B 3.8-59	0	B 3.9-2	0
B 3.8-60	0	B 3.9-3	68
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B 3.8-62	0	B 3.9-5	0
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		B 3.9-9	0

TECHNICAL SPECIFICATIONS BASES

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REVISIONS	ISSUED	SUBJECT
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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REVISIONS	ISSUED	SUBJECT
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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REVISIONS	ISSUED	SUBJECT
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 21and 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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REVISIONS	ISSUED	SUBJECT
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 μCi/gm to <0.265 μCi/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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REVISIONS	ISSUED	SUBJECT
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.

TECHNICAL SPECIFICATION BASES - REVISION LISTING
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REVISIONS	ISSUED	SUBJECT
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).

TECHNICAL SPECIFICATION BASES - REVISION LISTING

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REVISIONS	ISSUED	SUBJECT
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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REVISIONS	ISSUED	SUBJECT
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.

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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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REVISIONS	ISSUED	SUBJECT
Revision 103	04-05-2010	Bases Revision 103 implemented NRC guidance "Application of Generic Letter 80-30" which allows a departure from the single failure criterion where a non-TS support system has two 100% capacity subsystems, each capable of supporting the design heat load of the area containing the TS equipment.
Revision 104 (Amendment 82)	09-20-2010	Bases Revision 104 implemented License Amendment No. 82, which approved the BEACON-TSM application of the Power Distributing System. The PDMS requirements reside in the TRM.
Revision 105	10-28-2010	DCN 53437 added spare chargers 8-S and 9-S which increased the total of 125 VDC Vital Battery Chargers to eight (8).
Revision 106	01-20-2011	Revised SR 3.8.3.6 to clarify that identified fuel oil leakage does not constitute failure of the surveillance.
Revision 107 (Amendment 85)	02-24-2011	Amendment 85 revises TS 3.7.11, "Control Room Emergency Air Temperature Control System (CREATCS). Specifically, the proposed change will only be applicable during plant modifications to upgrade the CREATCS chillers. This "one-time" TS change will be implemented during Watts Bar Nuclear Plant, Unit 1 Cycles 10 and 11 beginning March 1, 2011, and ending April 30, 2012.
Revision 108	03-07-2011	Bases Revision 108 deletes reference to NSRB to be notified of violation of a safety limit within 24 hours in TSB 2.2.4. Also, corrected error in SR 3.3.2.4 in the reference to Table 3.3.1-1. It should be Table 3.3.2-1.
Revision 109	04-06-2011	Bases Revision 109 clarifies that during plant startup in Mode 2 the AFW anticipatory auto-start signal need not be OPERABLE if the AFW system is in service. PER 287712 was identified by NRC to provide clarification to TS Bases 3.3.2, Function 6.e, Trip of All Turbine Driven Main Feedwater Pumps.
Revision 110	04-19-2011	Clarified the text associated with the interconnection of the ABI and CVI functions in the bases for LCO 3.3.6, 3.3.8, 3.7.12 and 3.9.8.

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REVISIONS	ISSUED	SUBJECT
Revision 111	05-05-2011	Added text to several sections of the Bases for LCO 3.4.16 to clarify that the actual transient limit for I-131 is 14 μCi/gm and refers to the controls being placed in AOI-28.
Revision 112	05-24-2011	DCN 55076 replaces the existing four 125-Vdc DG Battery Chargers with four sets of redundant new battery charger assemblies.
Revision 113	06-24-2011	Final stage implementation of DCN 55076 which replaced the existing four 125-Vdc DG Battery Chargers with four sets of redundant new battery charger assemblies.
Revision 114	12-12-2011	Clarifies the acceptability of periodically using a portion of the 25% grace period in SR 3.0.2 to facilitate 13 week maintenance work schedules.
Revision 115	12-21-2011	Revises several surveillance requirements notes in TS 3.8.1 to allow performance of surveillances on WBN Unit 2 6.9 kV shutdown boards and associated diesel generators while WBN Unit 1 is operating in MODES 1, 2, 3, or 4
Revision 116	06-27-2012	Revises TS Bases 3.8.1, AC Sources - Operating, to make the TS Bases consistent with TS 3.8.1, Condition D
Revision 117	07-27-2012	Revises TS Bases 3.7.10, Control Room Emergency Ventilation System (CREVS), to make the TS Bases consistent with TS 3.7.10, Condition E
Revision 118	01-30-2013	Revises TS Bases 3.4.16, Reactor Coolant System (RCS) to change the dose equivalent I-131 spike limit and the allowable value for control room air intake radiation monitors.
Revision 119	08-17-2013	Revises TS Bases 3.3.6, 3.3.8, 3.7.12, 3.7.13, 3.9.4, 3.9.7, 3.9.8, and adds TS Bases 3.9.10 to reflect selective implementation of the Alternate Source Term methodology for the analysis of Fuel Handling Accidents (FHAs) and make TS Bases consistent with the revised FHA dose analysis.
Revision 120	01-23-2014	Revised the References to TS Bases 3.1.9, PHYSICS TESTS Exceptions - Mode1, to document NRC approval of WCAP 12472-P-

		A. Addendum 1-A and 4-A. Addendum 1-A approved the use of the Advance Nodal Code (ANC-Phoenix_ in the BEACON system as the neutronic code for measuring core power distribution. Is also approved the use of fixed incore self-powered neutron detectors (SPD_ to calibrate the BEACON system in lieu of incore and excore neutron detectors and core exit thermocouples (CET). For plants that do not have SPDs Addendum 4-A approved Westinghouse methodology that allow the BEACON system to calculate CET uncertainty as a function of reactor power on a plant cycle basis during power ascension following a refueling outage.
Revision 121	08-04-2014	Revises a references in TS Bases 3.7.1 for consistency with changes to the TS Bases 3.7.1 references approved in Revision 89.

ENCLOSURE 2
WBN TECHNICAL SPECIFICATION BASES
CHANGED PAGES

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.9.4 (continued)

f. Samarium concentration; and

g. Deleted

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

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident without the required SDM.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

2. Title 10, Code of Federal Regulations, Part 50.59, "Changes, Tests, and Experiments."

3. Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978.

4. ANSI/ANS-19.6.1, "Reload Startup PHYSICS TESTS for Pressurized Water Reactors," American National Standards Institute.

5. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.

6. Watts Bar FSAR, Section 14.2, "Test Program."

7. WCAP-11618, "MERITS Program - Phase II, Task 5, Criteria Application," dated November 1987, including Addendum 1, April 1989.

8. WCAP-12472-P-A,"BEACON Core Monitoring and Operations Support System," August 1994 (Addendum 1-A, January 2000, Addendum 4-A, September 2012).

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Vent Isolation Instrumentation

BASES

BACKGROUND

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

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

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

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

(continued)

APPLICABLE SAFETY ANALYSES

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

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

(continued)

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 Allowable Values (AVs) for the Containment Purge Exhaust Radiation Monitors. This AV is based on expected concentrations for a small break LOCA, which is more restrictive than the 10 CFR 100 limits. The specified AV is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the AV. 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. 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)

APPLICABILITY (continued)

While in MODES 5 and 6, 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

ACTIONS (continued)

B.1

Condition B applies to all Containment Vent Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation. A Note has been added above the Required Actions to allow one train of actuation logic to be placed in bypass and to delay entering the Required Actions for up to four hours to perform surveillance testing provided the other train is OPERABLE. The 4 hour allowance is consistent with the Required Actions for actuation logic trains in LCO 3.3.2, "Engineered Safety Features Actuation System Instrumentation" and allows periodic testing to be conducted while at power without causing an actual actuation. The delay for entering the Required Actions relieves the administrative burden of entering the Required Actions for isolation valves inoperable solely due to the performance of surveillance testing on the actuation logic and is acceptable based on the OPERABILITY of the opposite train.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Vent Isolation Functions.

SR 3.3.6.1

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.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8 Auxiliary Building Gas Treatment (ABGTS) Actuation Instrumentation

BASES

BACKGROUND

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

There are a total of two channels, one for each train. A Phase A isolation signal from the Engineered Safety Features Actuation System (ESFAS) initiates auxiliary building isolation and starts the ABGTS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the Auxiliary Building Secondary Containment Enclosure (ABSCE).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

(continued)

BASES (continued)

LCO

The LCO requirements ensure that instrumentation necessary to initiate the ABGTS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the ABGTS at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one hand switch and the interconnecting wiring to the actuation logic relays.

2. Deleted

(continued)

BASES

LCO (continued)

3. <u>Containment Phase A Isolation</u>

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

APPLICABILITY

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

While in MODES 5 and 6, the ABGTS instrumentation need not be OPERABLE. See additional discussion in the Background and Applicable Safety Analysis sections.

ACTIONS

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

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

(continued)

BASES

ACTIONS (continued)

A.1

Condition A applies to the actuation logic train function from the Phase A Isolation and the manual function. Condition A applies to the failure of a single actuation logic train or manual channel. If one channel or train is inoperable, a period of 7 days is allowed to restore it to OPERABLE status. If the train cannot be restored to OPERABLE status, one ABGTS train must be placed in operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as that provided in LCO 3.7.12.

B.1.1, B.1.2, B.2

Condition B applies to the failure of two ABGTS actuation logic signals from the Phase A Isolation or two manual channels. The Required Action is to place one ABGTS train in operation immediately. This accomplishes the actuation instrumentation function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.12 must also be entered for the ABGTS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed on train inoperability as discussed in the Bases for LCO 3.7.12.

Alternatively, both trains may be placed in the emergency radiation protection mode. This ensures the ABGTS Function is performed even in the presence of a single failure.

BASES

ACTIONS

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which ABGTS Actuation Functions.

SR 3.3.8.1

SR 3.3.8.1 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 relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

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

REFERENCES

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

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ACTIONS (continued)

A.1

In the case of only a single inoperable MSSV on one or more steam generators a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressuration in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1, requires an appropriate reduction in reactor power within 4 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined using a conservative heat balance between the reactor coolant system heat generation and the steam relief through the OPERABLE MSSVs, as shown below and described in the attachment to Reference 5:

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

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

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

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

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

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined using a conservative heat balance calculation as described above (Action A.1) and in the attachment to Reference 5. The values in Specification 3.7.1 include an allowance for instrument and channel uncertainties to the allowable RTP obtained with this algorithm.

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

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

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

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

BASES

BACKGROUND

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

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

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

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a LOCA. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The DBA analysis assumes that only one train of the ABGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the ABSCE is determined for a LOCA. The assumptions and analysis for a LOCA follow the guidance provided in Regulatory Guide 1.4 (Ref. 6).

The ABGTS satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

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

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

a. Fan is OPERABLE;

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

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

APPLICABILITY

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

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

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1 and B.2

When Required Action A.1 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. The system must be operated for ≥ 10 continuous hours with the heaters energized. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required ABGTS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ABGTS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 8). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

(continued)

REFERENCES (continued)

5. Deleted

6. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors."

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

8. Regulatory Guide 1.52 (Rev. 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."

9. NUREG-0800, Section 6.5.1, "Standard Review Plan," Rev. 2, "ESF Atmosphere Cleanup System," July 1981.

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

11. Deleted.

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section 15.4.5 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 6). The Total Effective Dose Equivalent (TEDE) for control room occupants, individuals at the exclusion area boundary, and individuals within the low population zone will remain within 10 CFR 50.67 (Ref. 7) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref. 6) for a fuel handling accident.

According to Reference 6, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 6 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

The fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 (continued)

pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

REFERENCES

1. Watts Bar FSAR, Section 9.1.2, "Spent Fuel Storage."
2. Watts Bar FSAR, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System."
3. Watts Bar FSAR, Section 15.4.5, "Fuel Handling Accident."
4. Deleted
5. Deleted
6. Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.
7. Title 10, Code of Federal Regulations, 10 CFR 50.67, “Accident Source Term.”

B 3.9 REFUELING OPERATIONS

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B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 2 and 8). Sufficient iodine activity would be retained to limit offsite doses from the accident to the limits defined in 10 CFR 50.67 (Ref. 7) and Regulatory Position C.4.4. of Regulatory Guide 1.183 (Ref. 8).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment (Refs. 2 and 8). A minimum water level of 23 ft (Regulatory Position 2 of Appendix B to Regulatory Guide 1.183 (Ref. 8)) allows an overall iodine decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the I-131, 10% of the Kr-85, and 5% of the other noble gases and iodides from the total fission product inventory in accordance with Regulatory Position 3.1 of Regulatory Guide 1.183 (Ref. 8).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft in conjunction with a minimum decay time of 100 hours prior to fuel handling without containment closure or Auxiliary Building isolation, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Refs. 7 and 8).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement.

(Continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

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

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

REFERENCES

1. Deleted.

2. Watts Bar FSAR, Section 15.4.5, "Fuel Handling Accident."

3. NUREG-0800, "Standard Review Plan," Section 15.7.4, "Radiological Consequences of Fuel-Handling Accidents," U.S. Nuclear Regulatory Commission.

4. Title 10, Code of Federal Regulations, Part 20.1201(a), (a)(1), and(2)(2), "Occupational Dose Limits for Adults."

5. Deleted

6. Deleted.

7. Title 10, Code of Federal Regulations, 10 CFR 50.67, “Accident Source Term.”

8. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

B 3.9 REFUELING OPERATIONS

B 3.9.8 Deleted

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B 3.9 REFUELING OPERATIONS

B 3.9.10 Decay Time

BASES

BACKGROUND

Section 15.5.6 of the WBN, Unit 1 UFSAR (Ref. 1) defines the assumptions of the fuel handling accident radiological analysis, including a minimum decay time for irradiated fuel assemblies prior to movement. This assumption ensures that the inventory of radioactive isotopes is at a level that supports the safety analysis assumptions.

To ensure that irradiated fuel assemblies have decayed for the appropriate period of time, a limitation is established to require the reactor core to be subcritical for a time period at least equivalent to the minimum decay time assumption in the fuel handling analysis prior to allowing irradiated fuel to be moved.

Given that no irradiated fuel assembly will be moved outside of the containment until the minimum decay time requirement is met, this requirement also ensures that any irradiated fuel assemblies that are moved outside of the containment meet the decay time assumption in the radiological analysis of the fuel handling accident.

APPLICABLE SAFETY ANALYSES

The radiological analysis of the fuel handling accident (Ref. 1) assumes a minimum decay time prior to movement of irradiated fuel assemblies. The requirements of LCO 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation," LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)," LCO 3.7.11, "Control Room Emergency Air Temperature Control System (CREATCS)," and LCO 3.9.7, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the requirements of 10 CFR 50.67 (Ref. 2) and Regulatory Position C.4.4 of Regulatory Guide 1.183 (Ref. 3).

The decay time satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO	A minimum decay time of 100 hours is required prior to moving irradiated fuel assemblies within containment. This preserves an assumption in the fuel handling accident analysis (Ref. 1), and ensures that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies within the containment, since the potential for a release of fission products exists.
ACTIONS	<u>A.1</u> When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the reactor is subcritical for < 100 hours, movement of irradiated fuel assemblies within containment must be suspended. This action precludes the possibility of a fuel handling accident in containment. This action does not preclude moving a fuel assembly to a safe position. The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.10.1</u> This SR verifies that the reactor has been subcritical for at least 100 hours prior to moving irradiated fuel assemblies by confirming the date and time of subcriticality. This ensures that any irradiated fuel assemblies have decayed for at least 100 hours prior to movement. The Frequency of "Prior to movement of irradiated fuel in the containment" is appropriate, because it ensures that the decay time requirement has been met just prior to moving the irradiated fuel.
REFERENCES	1. Watts Bar UFSAR, Section 15.5.6, "Environmental Consequences of a Postulated Fuel Handling Accident." 2. Title 10, Code of Federal Regulations, 10 CFR 50.67, "Accident Source Term." 3. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

ENCLOSURE 3
WBN TECHNICAL REQUIREMENTS MANUAL
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TECHNICAL REQUIREMENTS

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BASES

(continued)

LIST OF TABLES

LIST OF FIGURES

LIST OF MISCELLANEOUS REPORTS AND PROGRAMS

LIST OF ACRONYMS

Acronym	Title
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
ARV	Atmospheric Relief Valve
BOC	Beginning of Cycle
CCS	Component Cooling Water 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
LCC	Lower Compartment Cooler
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
PDMS	Power Distribution Monitoring System
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
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

TECHNICAL REQUIREMENTS

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Revisions	Issued	SUBJECT
Revision 0	09-30-95	Initial Issue
Revision 1	12-06-95	Submerged Component Circuit Protection
Revision 2	01-04-96	Area Temperature Monitoring - Change in MSSV Limit
Revision 3	02-28-96	Turbine Driven AFW Pump Suction Requirement
Revision 4	08-18-97	Time-frame for Snubber Visual Exams
Revision 5	08-29-97	Performance of Snubber Functional Tests at Power
Revision 6	09-08-97	Revised Actions for Turbine Overspeed Protection
Revision 7	09-12-97	Change OPΔT/OTΔT Response Time
Revision 8	09-22-97	Clarification of Surveillance Frequency for Position Indication System
Revision 9	10-10-97	Revised Boron Concentration for Borated Water Sources
Revision 10	12-17-98	ICS Inlet Door Position Monitoring - Channel Check
Revision 11	01-08-99	Computer-Based Analysis for Loose Parts Monitoring
Revision 12	01-15-99	Removal of Process Control Program from TRM
Revision 13	03-30-99	Deletion of Power Range Neutron Flux High Negative Rate Reactor Trip Function
Revision 14	04-07-99	Submerged Component Circuit Protection
Revision 15	04-07-99	Submerged Component Circuit Protection
Revision 16	04-13-99	Submerged Component Circuit Protection
Revision 17	05-25-99	Flood Protection Plan
Revision 18	08-03-99	Submerged Component Circuit Protection
Revision 19	10-12-99	Upgrade Seismic Monitoring Instruments
Revision 20	03/13/00	Added Notes to Address Instrument Error for Various Parameters
Revision 21	04/13/00	COLR, Cycle 3, Rev 2
Revision 22	07/07/00	Elimination of Response Time Testing

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Revisions	Issued	SUBJECT
Revision 23	01/22/01	Plant Calorimetric (LEFM)
Revision 24	03/19/01	TRM Change Control Program per 50.59 Rule
Revision 25	05/15/01	Change in Preventive Maintenance Frequency for Molded Case Circuit Breakers
Revision 26	05/29/01	Change CVI Response Time from 5 to 6 Seconds
Revision 27	01/31/02	Change pH value in the borated water sources due to TS change for ice weight reduction
Revision 28	02/05/02	Refueling machine upgrade under DCN D-50991-A
Revision 29	02/26/02	Added an additional action to TR 3.7.3 to perform an engineering evaluation of inoperable snubber's impact on the operability of a supported system.
Revision 30	06/05/02	Updated TR 3.3.5.1 to reflect implementation of the TIPTOP program in a Technical Instruction (TI).
Revision 31	10/31/02	Correct RTP to 3459 MWt (PER 02-9519-000)
Revision 32	09/17/03	Editorial correction to Bases for TSR 3.1.5.3.
Revision 33	10/14/03	Updated TRs 3.1.5 and 3.1.6 and their respective bases to incorporate boron concentration changes in accordance with change packages WBN-TS-02-14 and WBN-TS-03-017.
Revision 34	05/14/04	Revised Item 5, "Source Range, Neutron Flux" function of Table 3.3.1-1 to provide an acceptable response time of less than or equal 0.5 seconds. (Reference TS Amendment 52.)
Revision 35	04/06/05	Revised Table 3.3.2-1, "Engineered Safety Features Actuation systems Response Times," to revise containment spray response time and to add an asterisk note to notation 13 of the table via Change Package WBN-TS-04-16.
Revision 36	09/25/06	Revised the response time for Containment Spray in Table 3.3.2-1 and the RT_{NDT} values in the Bases for TR 3.7.1. Both changes result from the replacement of the steam generators.
Revision 37	11/08/06	Revised TR 3.1.5 and 3.1.6 and the Bases for these TRs to update the boron concentration limits of the RWST and the BAT.

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REVISION LISTING

Revisions	Issued	SUBJECT
Revision 38	11/29/06	Updated the TRM to be consistent with Tech Spec Amendment 55. TRM Revision 38 modified the requirements for mode change limitations in TR 3.0.4 and TSR 3.0.4 by incorporating changes similar to those outlined in TSTF-359, Revision 9. (TS-06-24)
Revision 39	04/16/07	Updated the TRM to be consistent with Tech Spec Amendment 42. TRM Revision 39 modified the requirements of TSR 3.0.3 by incorporating changes similar to those outlined in TSTF-358. (TS-07-03)
Revision 40	05/24/07	Updated the TRM and Bases to remove the various requirements for the submittal of reports to the NRC. (TS-07-06)
Revision 41	05/25/07	Revision 41 updates the Bases of TR 3.1.3, 3.1.4 and 3.4.5 to be consistent with Technical Specification Amendment 66. This amendment replaces the references to Section XI of the ASME Boiler and Pressure Vessel Code with the ASME Operation and Maintenance Code for Inservice Testing (IST) activities and removes reference to “applicable supports” from the IST program.
Revision 42	03/20/2008	Revision 42 updates Figure 3.1.6 to remove the 240 TPBAR Limit.
Revision 43	07/17/2008	Revision 43 removes a reporting requirement from TR 3.7.4, “Sealed Source Contamination.” The revision also updates the Bases for TR 3.7.4.
Revision 44	10/10/2008	Revision 44 updates Table 3.3.1-1 to be consistent with the changes approved by NRC as Tech Spec Amendment 68.
Revision 45	02/23/2009	Added TR 3.3.8, “Hydrogen Monitors,” and the Bases for TR 3.3.8. This change is based on Technical Specification (TS) Amendment 72 which removed the Hydrogen Monitors (Function 13 of LCO 3.3.3) from the TS.
Revision 46	09/20/2010	Revision 46 implements changes from License Amendment 82 (Technical Specification (TS) Bases Revsion 104) for the approved BEACON-TSM application of the Power Distribution Monitoring System (PDMS).
Revision 47	10/08/2010	Revision 47 changes are in response to PER 215552 which requested clarification be added to the TRM regarding supported system operability when a snubber is declared inoperable or removed from service.

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Revisions	Issued	SUBJECT
Revision 48	04/12/2011	CANCELLED
Revision 49	05/24/2011	Revision 49 updated Note 14 of Table 3.3.2-1 to clarify that the referenced time is only for 'partial' transfer of the ECCS pumps from the VCT to the RWST.
Revision 50	12/12/2011	Clarifies the acceptability of periodically using a portion of the 25% grace period in TSR 3.0.2 to facilitate 13 week maintenance work schedules.
Revision 51	08/09/2013	Adds a note to TR 3.1.2 and TR 3.1.4 to permit securing one charging pump in order to supporting transition into or from the Applicability of Technical Specification 3.4.12 (PER 593365).
Revision 52	08/30/2013	Clarifies that TR 3.4.5, "Piping System Structural Integrity," applies to all ASME Code Class 1, 2, and 3 piping systems, and is not limited to reactor coolant system piping.
Revision 53	12/12/2013	Technical Specification Amendment 92 added Limiting Condition for Operation (LCO) 3.9.10, "Decay Time," which was redundant to Technical Requirement (TR) 3.9.1, "Decay Time." Revision 53 removes TR 3.9.1 from the Technical Requirements Manual (TRM) and the TRM Bases.
Revision 54	01/14/2014	Update the references for B 3.3.9 to incorporate NRC approved WCAP-12472-P-A Addendum 1-A and 4-A. The WCAPs provide the bases to allow the core exit thermocouple uncertainty to be calculated on a cycle specific basis. The new method obtains thermocouple data during power ascension following a refueling outage. This information is processed by the PDMS to determine the power dependent thermocouple uncertainty.

ENCLOSURE 4
WBN TECHNICAL REQUIREMENTS MANUAL
CHANGED PAGES

TR 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.2 Boration Systems Flow Paths, Operating

TR 3.1.2 Two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid tanks, through a boric acid transfer pump, through a charging pump to the Reactor Coolant System (RCS).

b. Two flow paths from the Refueling Water Storage Tank (RWST), through charging pumps to the RCS.

-----NOTE-----
In MODE 3, a charging pump may be made incapable of injecting to support transition into or from the Applicability of the TS LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)," for up to four hours or until the temperature of all the RCS cold legs exceeds 375°F, whichever occurs first.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required flow path inoperable.	A.1 Restore required flow path to OPERABLE status.	72 hours
	<u>OR</u>	
	A.2.1 Be in MODE 3.	78 hours
	<u>AND</u>	
	A.2.2 Borate to a SDM equivalent to $\geq$ 1% $\Delta k/k$ at 200°F.	78 hours
	<u>AND</u>	
	A.2.3 Restore required path to OPERABLE status.	246 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	6 hours

TR 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.4 Charging Pumps, Operating

TR 3.1.4 Two charging pumps shall be OPERABLE.

-----NOTE-----
In MODE 3, a charging pump may be made incapable of injecting to support transition into or from the Applicability of the TS LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)," for up to four hours or until the temperature of all the RCS cold legs exceeds 375°F, whichever occurs first.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required charging pump inoperable.	A.1 Restore required charging pump to OPERABLE status.	72 hours
	<u>OR</u>	
	A.2.1 Be in MODE 3.	78 hours
	<u>AND</u>	
	A.2.2 Borate to a SDM equivalent to $\geq$ 1% $\Delta k/k$ at 200°F.	78 hours
	<u>AND</u>	
	A.2.3 Restore required charging pump to OPERABLE status.	246 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	6 hours

TR 3.4 REACTOR COOLANT SYSTEM (RCS

TR 3.4.5 Piping System Structural Integrity

TR 3.4.5 The structural integrity of ASME Code Class 1, 2, and 3 components in all systems shall be maintained in accordance with TSR 3.4.5.1 and TSR 3.4.5.2.

APPLICABILITY: All MODES.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Structural integrity of any ASME Code Class 1 component(s) not within limits.	A.1 Restore structural integrity of affected component(s) to within limit.	Prior to increasing Reactor Coolant System temperature > 50°F above the minimum temperature required by NDT considerations.
	OR	
	A.2 Isolate affected component(s).	Prior to increasing Reactor Coolant System temperature > 50°F above the minimum temperature required by NDT considerations.

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.5.1	Inspect each reactor coolant pump flywheel according to the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.	According to the recommend-actions of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1.
TSR 3.4.5.2	Verify the structural integrity of ASME Code Class 1, 2, and 3 components in all systems are in accordance with the Inservice Inspection Program.	In accordance with the Inservice Inspection Program.

TR 3.9 REFUELING OPERATIONS

TR 3.9.1 Deleted

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Boration Systems Flow Paths, Operating

BASES

BACKGROUND	A description of the Boration Systems Flow Paths is provided in the Bases for Technical Requirement 3.1.1, "Boration Systems Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or Transient. In the case of a malfunction of the Chemical and Volume Control System, which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1). OPERABILITY of the charging pumps, the Refueling Water Storage Tank (RWST), and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.
TR	TR 3.1.2 requires at least two boron injection flow paths to be OPERABLE during MODES 1, 2, and 3, in order to provide two redundant paths to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations. This requirement may be achieved by having two of the following three flow paths OPERABLE: a. One flow path from the boric acid storage tanks, through a boric acid transfer pump, through a charging pump to the Reactor Coolant System (RCS). b. Two flow paths from the RWST, through a charging pump to the RCS. TR 3.1.2 is modified by a Note. As indicated in this note, operation in MODE 3 with a charging pump made incapable of injecting in order to facilitate entry into or exit from the Applicability of TS LCO 3.4.12, “Cold Overpressure Mitigation Systems (COMS)” is necessary with a COMS arming temperature at or near the Mode 4 boundary temperature of 350°F. TS LCO 3.4.12 requires that certain pumps be rendered incapable of injecting at and below the COMS arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to make pumps incapable of injecting prior to entering the COMS Applicability, and provide time to restore the inoperable pumps to OPERABLE status on exiting the COMS Applicability.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Charging Pumps, Operating

BASES

BACKGROUND

A description of the Boration Systems Flow Paths is provided in the Bases for Technical Requirement 3.1.1, "Boration Systems Flow Paths, Shutdown."

APPLICABLE SAFETY ANALYSES

The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the Chemical and Volume Control System (CVCS), which causes a boron dilution event, the response required by the operator is to close the appropriate valves in the reactor makeup system and/or stop the primary water pumps. This action is required before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1). OPERABILITY of the charging pumps, the refueling water storage tank, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.

TR

TR 3.1.4 requires at least two charging pumps to be OPERABLE during MODES 1, 2, and 3 in order to assure redundant pumps to the two redundant flow paths to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations.

TR 3.1.2 is modified by a Note. As indicated in this note, operation in MODE 3 with a charging pump made incapable of injecting in order to facilitate entry into or exit from the Applicability of TS LCO 3.4.12, "Cold Overpressure Mitigation Systems (COMS)" is necessary with a COMS arming temperature at or near the Mode 4 boundary temperature of 350°F. TS LCO 3.4.12 requires that certain pumps be rendered incapable of injecting at and below the COMS arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to make pumps incapable of injecting prior to entering the COMS Applicability, and provide time to restore the inoperable pumps to OPERABLE status on exiting the COMS Applicability.

(continued)

BASES

B 3.3 INSTRUMENTATION

B 3.3.9 Power Distribution Monitoring System (PDMS)

BASES

BACKGROUND

The Power Distribution Monitoring System (PDMS) generates a continuous measurement of the incore power distribution using the methodology documented in References 1, 4, and 5. The PDMS employs an advanced three-dimensional nodal code to calculate the incore power distribution. The reference incore power distribution is periodically normalized to the incore flux measurements from the movable incore detectors. On a nominal once-per-minute basis, the incore power distribution is updated with plant instrumentation, most notably from the Core Exit Thermocouples (CETs). In this way, the information from the up-to-the-minute PDMS incore power distribution is equivalent to a full incore flux map using the Movable Incore Detector System (Technical Requirement 3.3.3).

The PDMS incore power distribution measurement can be used to determine the most limiting core peaking factors, $F^N_{\Delta H}$, the Nuclear Enthalpy Rise Hot Channel Factor (Technical Specification 3.2.2) and $F_Q(z)$, the Heat Flux Hot Channel Factor (Technical Specification 3.2.1). The incore power distribution measurement can also be used in the calibration of the excore neutron flux detection system (Technical Specification 3.3.1), monitoring the QUADRANT POWER TILT RATIO (QPTR) (Technical Specification 3.2.4), and verifying the position of a rod with inoperable position indicators (Technical Specification 3.1.8).

The PDMS requires information on current plant and core conditions in order to determine the core power distribution using the core peaking factor measurement and measurement uncertainty methodology described in References 1, 4, and 5. The OPERABILITY of the PDMS with the specified minimum complement of instrumentation channel inputs ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The PDMS requires input for power range detector calibrated voltage values, average reactor vessel inlet temperature, reactor power level, control bank positions, and temperatures from the CETs.

Either the PDMS or the Movable Incore Detector System may be used for calibration of the Excore Neutron Flux Detection System, monitoring the QUADRANT POWER TILT RATIO, or measurement of $F_Q(z)$ or $F^N_{\Delta H}$. Similarly, either the PDMS or the Movable Incore Detector System may be used for verifying the position of a rod with inoperable position indicators, but only the PDMS must satisfy OPERABILITY requirements prior to this function.

(continued)

Bases (continued)

APPLICABLE SAFETY ANALYSES

The PDMS is used for periodic measurement of the core power distribution to confirm operation within design limits and periodic calibration of the excore detectors. This system does not initiate any automatic protection action. The PDMS is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient (References 2 and 3).

TR

TR 3.3.9 requires the PDMS to be OPERABLE with the specified number of instrument channel inputs from the plant computer for each function listed in Table 3.3.9-1. The PDMS is OPERABLE when the required channel inputs are available, the calibration data set is valid, and reactor power is $\geq$ 25% RTP.

This TR ensures the OPERABILITY of the PDMS when required to monitor the power distribution within the core. The PDMS is used for periodic surveillance of the incore power distribution and calibration of the excore detectors. The surveillance of incore power distribution verifies that the peaking factors are within their design envelope (Reference 3). The peaking factor limits include measurement uncertainty which bounds the actual measurement uncertainty of an OPERABLE PDMS (References 1, 4, and 5).

Maintaining the minimum number of instrumentation channel inputs ensures the uncertainty is bounded by the uncertainty methodology. Similarly, when THERMAL POWER is less than 25% RTP, then the accuracy of the adjustment provided by the CETs to the measured PDMS power distribution may not be bounded by the uncertainties documented in .References 1, 4, and 5.

APPLICABILITY

The PDMS must be OPERABLE when it is used for calibration of the Excore Neutron Flux Detection System, monitoring the QPTR, measurement of $F^N_{\Delta H}$ and $F_Q(z)$, or verifying the position of a rod with inoperable position indicators.

ACTIONS

A.1

The Required Action A.1 has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

With THERMAL Power less than 25% RTP or with one or more required channel inputs inoperable or unavailable to the PDMS, the PDMS must not be used to obtain an incore power distribution measurement. Therefore, the Required Action A.1 prohibits the use of the inoperable system for the applicable monitoring or calibration functions.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

The subsequent PDMS calibration frequency is 31 Effective Full Power Days (EFPD) when the CET chess knight move pattern is not satisfied. The CET chess knight move pattern is satisfied when every interior core location (fuel assemblies not face adjacent to the core baffle) is no further than a chess knight's move from an OPERABLE CETC. The 31 EFPD frequency calibration requirement is modified by a note that clarifies that subsequent PDMS calibration is not required to be performed until 31 EFPD after the CET chess knight move pattern is not satisfied.

The subsequent PDMS calibration frequency is 180 EFPD when the CET chess knight move pattern is satisfied. The CET chess knight move pattern provides coverage of all interior fuel assemblies (coverage of fuel assemblies with a face along the baffle is not required). Fuel assemblies which are within a chess knight's move of an OPERABLE CET are covered.

REFERENCES

1. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

2. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

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

4. WCAP-12472-P-A, Addendum 1-A, "BEACON Core Monitoring and Operations Support System," January 2000.

5. WCAP-12472-P-A, Addendum 4-A, "BEACON Core Monitoring and Operations Support System," September 2012.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 Piping System Structural Integrity

BASES

BACKGROUND	Inservice inspection of ASME Code Class 1, 2, and 3 components and pressure testing of ASME Code Class 1, 2, and 3 pumps and valves in all systems are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (Ref. 1) and applicable Addenda, as required by 10 CFR 50.55a(g) (Ref. 2). Exception to these requirements apply where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i) and (a)(3). In general, the surveillance intervals specified in Section XI of the ASME Code apply. However, the Inservice Inspection Program includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Code. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications. Each reactor coolant pump flywheel is, in addition, inspected as recommended in Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 (Ref. 3). Additionally, programmatic information on Inservice Inspection is provided in Technical Specifications, Chapter 5.0, Administrative Controls, Section 5.7.2.11, Inservice Inspection Program.
APPLICABLE SAFETY ANALYSES	Certain components which are designed and manufactured to the requirements of specific sections of the ASME Boiler and Pressure Vessel Code are part of the primary success path and function to mitigate DBAs and transients. However, the operability of these components is addressed in the relevant specifications that cover individual components. In addition, this particular Requirement covers only structural integrity inspection/testing requirements for these components, which is not a consideration in designing the accident sequences for theoretical hazard evaluation (Ref.4).
TR	TR 3.4.5 requires that the structural integrity of the ASME Code Class 1, 2, and 3 components in all systems be maintained in accordance with TSR 3.4.5.1 and TSR 3.4.5.2. In those areas where conflict may exist between the Technical Specifications and the ASME Boiler and Pressure Vessel Code, the Technical Specifications take precedence.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.5.1

This surveillance stipulates inspection of the coolant pump flywheel in accordance with Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1. This inspection verifies the structural integrity of the flywheel.

TSR 3.4.5.2

TSR 3.4.5.2 requires the verification of structural integrity of ASME Code Class 1, 2, and 3 components in all systems are in accordance with the Inservice Inspection Program.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI.
2. 10 CFR 50.55a, "Codes and Standards."
3. Regulatory Guide 1.14, Revision 1, 1975.
4. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.9 REFUELING OPERATIONS

B 3.9.1 Deleted

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