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

WBL-19-023

March 19, 2019

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, Units 1 and 2
Facility Operating License Nos. NPF-90 and NPF-96
NRC Docket Nos. 50-390 and 50-391

Subject: Watts Bar Nuclear Plant Units 1 and 2 – Periodic Submission for Changes Made to the WBN Technical Specification Bases and Technical Requirements Manual

References: TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Periodic Submission for Changes Made to the WBN Technical Specification Bases and Technical Requirements Manual," dated November 2, 2017 (ML17306A802)

The purpose of this letter is to provide the Nuclear Regulatory Commission (NRC) with copies of changes to the Watts Bar Nuclear Plant (WBN) Units 1 and 2 Technical Specification (TS) Bases and to provide copies of changes to the Unit 1 and 2 Technical Requirements Manual (TRM). These changes have been implemented at WBN during the period since the last update (Reference 1). Copies of the TS Bases, Revisions 139 through 149 for Unit 1 and Revisions 12 through 18 for Unit 2, are provided in accordance with WBN Units 1 and 2 TS Section 5.6, "Technical Specifications (TS) Bases Control Program." In addition, copies of changes to the WBN Units 1 and 2 TRM, Revisions 65 and 66 for Unit 1 and Revisions 8 through 10 for Unit 2, are provided in accordance with WBN TRM Section 5.1, "Technical Requirements (TR) Control Program."

The changes 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 Title 10 of the Code of Federal Regulations (10 CFR) 50.71(e). 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. Should you have questions regarding this submittal, please contact Kim Hulvey, Site Licensing Manager, at (423) 365-7720.

Respectfully,

Anthony L. Williams IV
Site Vice President
Watts Bar Nuclear Plant

Enclosures:
1 - WBN Unit 1 Technical Specification Bases - Table of Contents
2 - WBN Unit 1 Technical Specifications Bases - Changed Pages
3 - WBN Unit 1 Technical Requirements Manual - Table of Contents
4 - WBN Unit 1 Technical Requirements Manual - Changed Pages
5 - WBN Unit 2 Technical Specification Bases - Table of Contents
6 - WBN Unit 2 Technical Specifications Bases - Changed Pages
7 - WBN Unit 2 Technical Requirements Manual - Table of Contents
8 - WBN Unit 2 Technical Requirements Manual - Changed Pages

cc (Enclosures):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector
NRR Project Manager

ENCLOSURE 1
WBN UNIT 1 TECHNICAL SPECIFICATION BASES
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LIST OF ACRONYMS

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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

LIST OF ACRONYMS

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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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.

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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).

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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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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.

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REVISIONS	ISSUED	SUBJECT
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 references in TS Bases 3.7.1 for consistency with changes to the TS Bases 3.7.1 references approved in Revision 89.
Revision 122 (Amendment 94)	01-14-2014	Revises TS Bases 3.7.10, Control Room Emergency Ventilation System (CREVS) to make the TS Bases consistent with TS 3.7.10, Actions E, F, G, and H.
Revision 123 (Amendment 104)	03-16-2016	Amendment 104, TSB Revision 123 adds TS B3.7.16, "Component Cooling System (CCS) - Shutdown" and adds TS B3.7.17, "Essential Raw Cooling Water (ERCW) System - Shutdown.
Revision 124	02-12-2016	Revises TS Bases Table B3.8.9-1, "AC and DC Electrical Power Distribution Systems," the second Note.
Revision 125 (Amendment 84, 102, 103)	03-16-2016	Revises TS Bases Section B3.8-1, "AC Sources-Operating."
Revision 126	03-18-2016	Revises TS Bases Section B3.7.7, "Component Cooling System" the 1B and 2B surge tank sections.
Revision 127	04-18-2016	Revises TS Bases Section B 3.6.4, "Containment Pressure" and B3.6.6, "Containment Spray System to change the maximum peak pressure from a LOCA of 9.36 psig.

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REVISIONS	ISSUED	SUBJECT
Revision 128	06-27-16	Revises TS Bases Section B3.6.8, "Hydrogen Mitigation System (HMS)", to delete sentence regarding Hydrogen Recombiners that are abandoned.
Revision 129	08-19-16	Revises TS Bases Section 3.6.15, "Shield Building," to clarify the use of the Condition B note.
Revision 130	12-22-16	Revises TS Bases Sections 3.6.1, 3.6.2, and 3.6.3 to reflect the deletion of TS 3.9.4 in WBN Unit 1 TS Amendment 92.
Revision 131 (Amendment 107)	01-13-17	Revises TS Bases Section 3.5.4, " Refueling Water Storage Tank (RWST), Applicable Safety Analyses"
Revision 132 (Amendment 110)	01-17-17	Revises TS Bases Section 3.8.1, "AC Sources -Operating"
Revision 133 (Amendment 111)	03-13-17	Adds TS Bases Section 3.0.8 for Inoperability of Snubbers.
Revision 134 (Amendment 112)	04-25-17	Revise TS Bases Section 3.7.11 Action A.1 regarding CREATCS.
Revision 135	05-17-17	Revises TS Bases Section B3.3.3, "PAM Instrumentation"
Revision 136 (Amendment 113)	05-17-17	Revises TS Bases Section B3.7.7 "CCS"
Revision 137 (Amendment 114)	07-14-17	Revises TS Bases Section B SR 3.0.2 to add a one-time extension for the surveillance interval.
Revision 138 (Amendment 115)	11-2-17	Revises TS Bases to adopt the TSTF-522 to revise ventilation system surveillance requirements to operate for 10 hours per month.
Revision 139 (Amendment 116)	11-2-17	Revises TS Bases Auxiliary Building Gas Treatment System.
Revision 140	12-12-17	Revises TS Bases to include the ABB-NV and WLOP secondary CHF correlations.
Revision 141	03-08-18	Revises TS Bases 3.0.6 to remove non-standard guidance added by Bases Rev.103 that applied LCO 3.0.6 to non-TS support equipment when the equipment consisted of two 100% capacity subsystems, each capable of supporting both trains of TS equipment.

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 142	04-06-18	Add clarifying information of ECCS gas that some gas is acceptable based on output of DCP 66453.
Revision 143, Amendment 120	08-20-18	Revises TS Bases and adopts the TSTF-547, Clarification of Rod position requirements.
Revision 144, Amendment 121	08-16-18	Revises TS Bases 3.0 to extend surveillance requirements and specify intervals.
Revision 145, Amendment 122	09-21-18	Revises TS Bases 3.2.4 and Bases 3.3.1 related to the reactor trip system instrumentation.
Revision 146, Amendment 119	10-11-18	Revises TS Bases 3.3.1 "Reactor Trip System Instrumentation," to reflect plant modifications to the reactor protection system instrumentation associated with the turbine trip on low fluid oil pressure.
Revision 147	11-14-18	Revises TS Bases 3.7.5, AFW System, to increase margin on the AFW MDAFW pumps.
Revision 148	11-14-18	Revises TS Bases 3.4.12, References section to update Reference 4 with an updated FSAR Section.
Revision 149	2-06-19	Revises TS Bases 3.3.1, "Reactor Trip System Instrumentation"

ENCLOSURE 2
WBN UNIT 1 TECHNICAL SPECIFICATION BASES
CHANGED PAGES

BASES

BACKGROUND (continued)

DNB is not a directly measurable parameter during operation; therefore, THERMAL POWER, reactor coolant temperature, and pressure are related to DNB through critical heat flux (CHF) correlations. The primary DNB correlations are the WRB-1 correlation (Ref. 7) for VANTAGE 5H and VANTAGE+ fuel and the WRB-2M correlation (Ref.8) for RFA-2 fuel with IFMs. These DNB correlations take credit for significant improvement in the accuracy of the CHF predictions. The W-3, ABB-NV, or WLOP CHF correlations (Refs. 9, 10, and 11) are used for conditions outside the range of the WRB-1 correlation for VANTAGE 5H and VANTAGE+ fuel or the WRB-2M correlation for RFA-2 fuel with IFMs.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Overtemperature ΔT trip

(continued)

BASES

References (continued)

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

5. Title 10, Code of Federal Regulations, Part 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors."

6. Title 10, Code of Federal Regulations, Part 50.73, "Licensee Event Report System."

7. WCAP-8762-P-A, "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," July 1984.

8. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

9. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, TID-25887, 1972.

10. Tong, L. S., "Critical Heat Fluxes on Rod Bundles," in "Two-Phase Flow and Heat Transfer in Rod Bundles," Engineers, New pages 31 through 41, American Society of Mechanical York, 1969.

11. WCAP-14565-P-A Addendum 2-P-A (Proprietary) / WCAP-15306-NP-A Addendum 2-NP-A (Non-Proprietary), "Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation WLOP for PWR Low Pressure Applications," April 2008.

BASES

LCO 3.0.6 (continued)

the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

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

Specification 5.7.2.18, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

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

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BASES (continued)

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities). For each of the SRs listed in Table SR 3.0.2-1 the specified Frequency is met if the Surveillance is performed on or before the date listed on Table SR 3.0.2-1. The Surveillance Frequency extension limits expire on the dates listed in Table SR 3.0.2-1 or when the unit enters MODE 5, whichever occurs first.

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the surveillance requirement will include a note in the frequency stating, "SR 3.0.2 does not apply." An example of an exception when the test interval is not specified in the regulations, is the discussion in the Containment Leakage Rate Testing Program, that SR 3.0.2 does not apply. This exception is provided because the program already includes extension of test intervals.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified, with the exception of surveillances required to be performed on a 31-day frequency. For surveillances performed on a 31-day frequency, the normal surveillance interval may be extended in accordance with Specification 3.0.2 cyclically as required to remain synchronized to the 13-week maintenance work schedules. This practice is acceptable based on the results of an evaluation of 31-day frequency surveillance test histories that demonstrate that no adverse failure rate changes have occurred nor would be expected to develop as a result of cyclical use of surveillance interval extensions and the fact that the total number of 31-day frequency surveillances performed in any one-year period remains unchanged.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy and Capability," (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each). A group consists

(continued)

BASES

BACKGROUND (continued)

of two or more RCCAs that are electrically paralleled to step simultaneously. Except for Shutdown Banks C and D, a bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Rod Position Indication (RPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The RPI System provides an accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six steps. The normal

(continued)

BASES

BACKGROUND (continued)

indication accuracy of the RPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ±12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

   1. Specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients other than a main steam line break (MSLB).

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Three types of analysis are performed in regard to static rod misalignment (Ref. 4). The first type of analysis considers the case where any one rod is completely inserted into the core with all other rods completely withdrawn. With control banks at their insertion limits, the second type of analysis considers the case when any one rod is completely inserted into the core. The third type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in each of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip in response to a main steam pipe rupture and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5). The reactor is shutdown by the boric acid injection delivered by the ECCS.

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly using incore power distribution measurements. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are acceptable from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

(continued)

BASES

LCO (continued)	Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.
APPLICABILITY	The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - T_{avg} > 200°F," for SDM in MODES 3 and 4, LCO 3.1.2, "Shutdown Margin (SDM)-T_{avg} ≤ 200°F" for SDM in MODE 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.
ACTIONS	<u>A.1.1 and A.1.2</u> When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating boration to restore SDM. In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

(continued)

ACTIONS (continued)

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1.1 and B.1.2

When a rod becomes misaligned, it can usually be moved and is still trippable.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7, "Control Bank Insertion Limits."

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

(continued)

ACTIONS

B.1.1 and B.1.2 (continued)

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2, B.3, B.4, and B.5

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 6). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$, and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain an incore power distribution measurement and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in UFSAR Change 15 (Ref. 3) that may be adversely affected will be evaluated to ensure that the analyses remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

(continued)

BASES

ACTIONS (continued)

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable.

To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the position of individual rod is within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

The SR is modified by a Note that permits it to not be performed for rods associated with an inoperable demand position indicator or an inoperable rod position indicator. The alignment limit is based on the demand position indicator which is not available if the indicator is inoperable. LCO 3.1.8, "Rod Position Indication," provides Actions to verify the rods are in alignment when one or more rod position indicators are inoperable.

The Surveillance is modified by a Note which states that the SR is not required to be performed until 1 hour after associated rod motion. Control rod temperature affects the accuracy of the rod position indication system. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows control rod temperature to stabilize following rod movement in order to ensure the indicated rod position is accurate.

SR 3.1.5.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.5.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.5.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after each reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 551°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

BACKGROUND (continued)

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are then left in this position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.7, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - T_{avg} > 200°F," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤ 200°F") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There be no violations of:

    1. Specified acceptable fuel design limits, or

    2. RCS pressure boundary integrity; and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. The core remains subcritical after accident transients other than a main steam line break (MSLB).

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

The LCO is modified by a Note indicating the LCO requirement is not applicable to shutdown banks being inserted while performing SR 3.1.3.2. This SR verifies the freedom of the rods to move, and may require the shutdown bank to move below the LCO limits, which would normally violate the LCO. This NOTE applies to each shutdown bank as it is moved below the insertion limit to perform the SR. This Note is not applicable should a malfunction stop performance of the SR.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. Refer to LCO 3.1.1 and LCO 3.1.2 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

(continued)

ACTIONS

A.1, A.2.1, A.2.2, and A.3

If one shutdown bank is inserted less than or equal to 10 steps below the insertion limit, 24 hours is allowed to restore the shutdown bank to within the limit. This is necessary because the available SDM may be reduced with a shutdown bank not within its insertion limit. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If a shutdown bank is not within its insertion limit, SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

While the shutdown bank is outside the insertion limit, all control banks must be within their insertion limits to ensure sufficient shutdown margin is available. The 24 hour Completion Time is sufficient to repair most rod control failures that would prevent movement of a shutdown bank.

B.1.1, B.1.2 and B.2

When one or more shutdown banks is not within insertion limits for reasons other than Condition A, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

C.1

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

The Surveillance is modified by a Note which states that the SR is not required to be performed for shutdown banks until 1 hour after motion of rods in those banks. Rod temperature affects the accuracy of the rod position indication system. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows rod temperature to stabilize following rod movement in order to ensure the indicated position is accurate.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," General Design Criterion 26, "Reactivity Control System Redundancy and Capability," and General Design Criterion 28, "Reactivity Limits."

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

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

BASES

BACKGROUND (continued)

to move with bank C on a withdrawal, as an example may be at 116 steps. Therefore, in this example, control bank C overlaps control bank D from 116 steps to the fully withdrawn position for control bank C. The fully withdrawn position and predetermined overlap positions are defined in the COLR.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.5, "Rod Group Alignment Limits," LCO 3.1.6, "Shutdown Bank Insertion Limits," LCO 3.1.7, "Control Bank Insertion Limits, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

   1. Specified acceptable fuel design limits, or

   2. Reactor Coolant System pressure boundary integrity; and

b. The core remains subcritical after accident transients other than a main steam line break (MSLB).

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3 through 13).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 5, 6, 8 and 11).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3 through 13).

The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analysis.

(continued)

BASES

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

The LCO is modified by a Note indicating the LCO requirement is not applicable to control banks being inserted while performing SR 3.1.5.2. This SR verifies the freedom of the rods to move, and may require the control bank to move below the LCO limits, which would normally violate the LCO. This Note applies to each control bank as it is moved below the insertion limit to perform the SR. This Note is not applicable should a malfunction stop performance of the SR.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

ACTIONS

A.1, A.2.1, A.2.2, and A.3

If Control Bank A, B, or C is inserted less than or equal to 10 steps below the insertion, sequence, or overlap limits, 24 hours is allowed to restore the control bank to within the limits. Verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If a control bank is not within its insertion limit, SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

While the control bank is outside the insertion, sequence, or overlap limits, all shutdown banks must be within their insertion limits to ensure sufficient shutdown margin is available and that power distribution is controlled. The 24 hour Completion Time is sufficient to repair most rod control failures that would prevent movement of a shutdown bank.

Condition A is limited to Control Banks A, B, or C. The allowance is not required for Control Bank D because the full power bank insertion limit can be met during performance of the SR 3.1.5.2 control rod freedom of movement (trippability) testing.

(continued)

BASES

ACTIONS (continued)

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

When the control banks are outside the acceptable insertion limits for reasons other than Condition A, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - T_{avg} > 200°F") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration for reasons other than Condition A, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

D.1

If the Required Actions cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with k_{eff} < 1.0, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.7.2

With an OPERABLE bank insertion limit monitor, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

The Surveillance is modified by a Note stating that the SR is not required to be performed for control banks until 1 hour after motion of rods in those banks. Control rod temperature affects the accuracy of the rod position indication system. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows control rod temperature to stabilize following rod movement in order to ensure the indicated rod position is accurate.

SR 3.1.7.3

When control banks are maintained within their insertion limits as checked by SR 3.1.7.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR.

The Surveillance is modified by a Note stating that the SR is not required to be performed for control banks until 1 hour after motion of rods in those banks. Control rod temperature affects the accuracy of the rod position indication system. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows control rod temperature to stabilize following rod movement in order to ensure the indicated rod position is accurate.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.3 (continued)

A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.7.2.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," General Design Criterion 26, "Reactivity Control System Redundancy and Capability," and General Design Criterion 28, "Reactivity Limits."

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

3. Watts Bar FSAR, Section 15.2.1, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition."

4. Watts Bar FSAR, Section 15.2.2, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power."

5. Watts Bar FSAR, Section 15.2.3, "Rod Cluster Control Assembly Misalignment."

6. Watts Bar FSAR, Section 15.2.4, "Uncontrolled Boron Dilution."

7. Watts Bar FSAR, Section 15.2.5, "Partial Loss of Forced Reactor Coolant Flow."

8. Watts Bar FSAR, Section 15.2.13, "Accidental Depressurization of the Main Steam System."

9. Watts Bar FSAR, Section 15.3.4, "Complete Loss of Forced Reactor Coolant Flow."

10. Watts Bar FSAR, Section 15.3.6, "Single Rod Cluster Control Assembly Withdrawal At Full Power."

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

12. Watts Bar FSAR, Section 15.4.4, "Single Reactor Coolant Pump Locked Rotor."

13. Watts Bar FSAR, Section 15.4.6, "Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)."

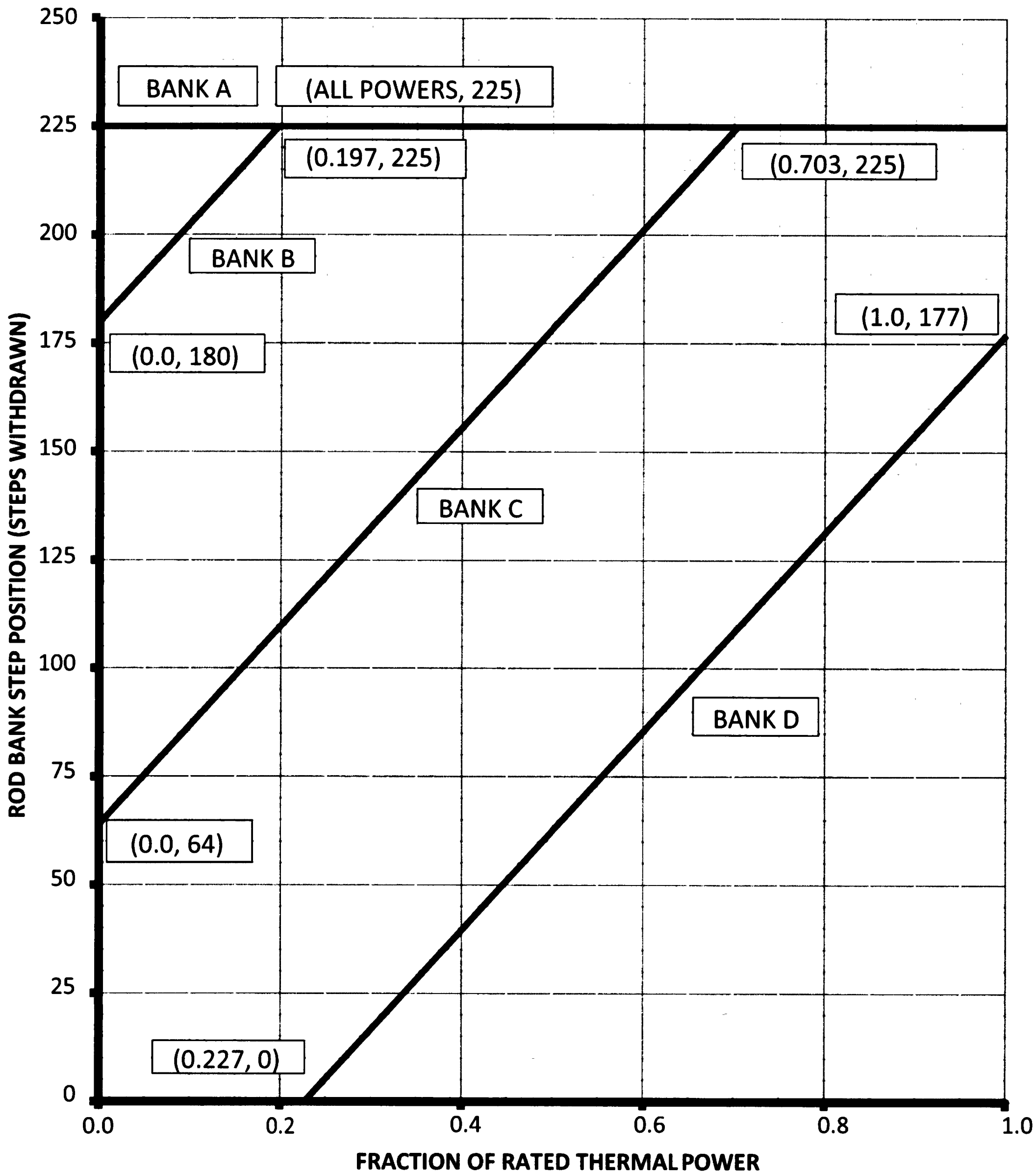

FIGURE B 3.1.7-1
CONTROL BANK INSERTION vs. RTP

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.8 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each).

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the analog Rod Position Indication (RPI) System.

(continued)

BASES

BACKGROUND (continued)

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The RPI System provides an accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of alternating primary and secondary coils spaced along a hollow tube. The normal indication accuracy of the RPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2 through 12), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.5, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.8 specifies that the RPI System and the Bank Demand Position Indication System be OPERABLE for all control rods. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO (when required) and the following:

(continued)

BASES

LCO (continued)

a. The RPI System indicates within 12 steps of the group step counter demand position when LCO 3.1.5, "Rod Group Alignment Limits;" met.

b. For the RPI System there are no failed coils; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the RPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the RPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.5, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

The LCO is modified by a Note stating that the RPI system is not required to be OPERABLE for 1 hour following movement of the associated rods. Control and shutdown rod temperature affects the accuracy of the RPI System. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows temperature to stabilize following rod movement in order to ensure the indicated position is accurate.

APPLICABILITY

The requirements on the RPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

(continued)

BASES

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1 and A.2

When one RPI channel per group in one or more groups fails, the position of the rod can still be determined indirectly by use of incore power distribution measurement information. Incore power distribution measurement information can be obtained from flux traces using the Movable Incore Detector System or from an OPERABLE Power Distribution Monitoring System (PDMS) (Ref. 15). The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the non-indicating rods have not been moved. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

Required Action A.1 requires verification of the position of a rod with an inoperable RPI once per 8 hours which may put excessive wear and tear on the moveable incore detector system. Required Action A.2 provides an alternative. Required Action A.2 requires verification of rod position using the incore power distribution measurement information every 31 EFPD, which coincides with the normal measurements to verify core power distribution.

Required Action A.2 includes six distinct requirements for verification of the position of rods associated with an inoperable RPI using incore power distribution measurements information:

a. Initial verification within 8 hours of the inoperability of the RPI;

b. Re-verification once every 31 Effective Full Power Days (EFPD) thereafter;

c. Verification within 8 hours after discovery of each unintended rod movement. An unintended rod movement is defined as the release of the rod's stationary gripper when no action was demanded either manually or automatically from the rod control system, or a rod motion in a direction other than the direction demanded by the rod control system. Verifying that no unintended rod movement has occurred is performed by monitoring the rod control system stationary gripper coil current for indications of rod movement;

(continued)

ACTIONS (continued)

d. Verification within 8 hours if the rod with an inoperable RPI is intentionally moved greater than 12 steps;

e. Verification prior to exceeding 50% RTP if power is reduced below 50% RTP; and

f. Verification within 8 hours of reaching 100% RTP if power is reduced to less than 100% power RTP.

Should the rod with the inoperable RPI be moved more than 12 steps, or if reactor power is changed, the position of the rod with the inoperable RPI must be verified.

A.3

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 4). The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Actions A.1 and A.2 above.

B.1 and B.2

When more than one RPI per group in one or more groups fail, additional actions are necessary. Placing the Rod Control System in manual assures unplanned rod motion will not occur. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition.

The inoperable RPIs must be restored, such that a maximum of one RPI per group is inoperable, within 24 hours. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the RPI system to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

C.1 and C.2

With one or more RPI inoperable in one or more groups and the affected groups have moved greater than 24 steps in one direction since the last determination of rod position, additional actions are needed to verify the position of rods with inoperable RPI. Within 4 hours, the position of the rods with inoperable position indication must be determined using either the moveable incore detectors or PDMS to verify these rods are still properly positioned, relative to their group positions.

(continued)

ACTIONS

C.1 and C.2 (continued)

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

D.1.1 and D.1.2

With one or more demand position indicators per bank inoperable in one or more banks, the rod positions can be determined by the RPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

D.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 13). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power to ≤ 50% RTP.

E.1

If the Required Actions cannot be completed within the associated Completion Time, 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. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the RPI agrees with the demand position within 12 steps ensures that the RPI is operating correctly.

This Surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

The Surveillance is modified by a Note which states it is not required to be met for RPIs associated with rods that do not meet LCO 3.1.5. If a rod is known to not be within 12 steps of the group demand position, the ACTIONS of LCO 3.1.5 provide the appropriate Actions.

(continued)

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;

b. During a loss of coolant accident (LOCA), the peak cladding temperature (PCT) must not exceed 2200°F for small breaks, and there must be a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 3);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum local DNB heat flux ratio to a value which satisfies the 95/95 criterion for the DNB correlation used. Refer to the Bases for the Reactor Core Safety Limits, B 2.1.1 for a discussion of the applicable DNBR limits. The W-3 Correlation with a DNBR limit of 1.3, or the ABB-NV correlation with a DNBR limit of 1.13, is applied in the heated region below the first mixing vane grid. In addition, the W-3 or WLOP DNB correlations are applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation for VANTAGE 5H and VANTAGE+ fuel or the WRB-2M correlation for RFA-2 fuel with IFMs. For system pressures in the range of 500 to 1000 psia, the W-3 correlation DNBR limit is 1.45 instead of 1.3. For system pressures in the range of 185 to 1800 psia, the WLOP correlation DNBR limit is 1.18.

(continued)

BASES

ACTIONS (continued)

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one power range neutron flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1 if more than one input from power range neutron flux channels are inoperable.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those changes of QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states the surveillance is only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels are inoperable when THERMAL POWER is > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2 (continued)

For the purpose of monitoring the QPTR when the input from one or more power range neutron flux channels are inoperable, incore power distribution measurement information is used to confirm that the indicated QPTR is consistent with the reference normalized symmetric power distribution. The incore power distribution information can be used to generate an incore "tilt." This tilt can be compared to the reference incore tilt to generate and incore QPTR. Therefore, incore QPTR can be used to confirm that excore QPTR is within limits.

The incore power distribution measurement information can be obtained from either the movable incore detectors or from an OPERABLE Power Distribution Monitoring System (PDMS) (Ref. 4). If the movable incore detectors are used, then the incore detector monitoring is performed with a full core flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-11, and N-8.

The reference normalized symmetric power distribution is available from the last incore power distribution measurement information used to calibrate the excore axial offset. The reference incore power distribution measurement information may have been obtained from either a full core flux map using the Movable Incore Detector System or from an OPERABLE PDMS. The full core flux map information may be reduced to the information from only the two sets of four symmetric thimbles with quarter core symmetry for like comparisons, if practical.

With the input from one or more power range neutron flux channels inoperable, the indicated QPTR may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might causes the QPTR limit to be exceeded, the normalized quadrant tilt is compared against the reference normalized quadrant tilt. Nominally, quadrant tilt from the surveillance should be within 2% of the tilt shown by the reference incore power distribution measurement information.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides backup protection to the Power Range Neutron Flux—Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. This Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip.

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

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

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

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA rod bank withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux—Low Setpoint and Intermediate Range Neutron Flux trip Functions. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. The LCO also requires one channel of the Source Range Neutron Flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux—Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS Source Range Neutron Flux trip Function may be manually blocked. Above the P-10 setpoint, the NIS Source Range Neutron Flux trip function is automatically blocked.

(continued)

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

a. Turbine Trip-Low Fluid Oil Pressure (continued)

power, will not actuate a reactor trip. Three pressure switches monitor the Emergency Trip Header pressure in the Turbine Electrohydraulic Control System high pressure header. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure—High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip — Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip—Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip—Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level below the P-9 setpoint, approximately 50% power. This action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure—High trip Function, and RCS integrity is ensured by the

(continued)

ACTIONS (continued)

D.1 and D.2

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 by Required Action D.1 to place the inoperable channel in the tripped condition is justified in Reference 14.

The Required Actions have been modified by two Notes. Note 1 allows the inoperable channel to be placed in the bypassed condition for up to 12 hours while performing routine surveillance testing of other channels. With one channel inoperable, the Note also allows routine surveillance testing of another channel with the inoperable channel in bypass. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the Power Range Neutron Flux-High setpoint in accordance with other Technical Specifications. The 12 hour time limit is justified in Reference 14.

Note 2 states to perform SR 3.2.4.2 if input to QPTR from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP.

If Required Action D.1 cannot be met within the specified Completion Time, 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 6 hours for the MODE reduction as required by Required Action D.2. 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.

(continued)

ACTIONS (continued)

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)

SURVEILLANCE REQUIREMENTS

SR 3.4.12.7 (continued)

The 12 hour allowance to meet the requirement considers the unlikelihood of a low temperature overpressure event during this time.

A Note has been added indicating that this SR is required to be met within 12 hours after decreasing RCS cold leg temperature to ≤ 350EF.

SR 3.4.12.8

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation."

3. ASME Boiler and Pressure Vessel Code, Section III.

4. Watts Bar FSAR, Section 5.2.2.4, "RCS Pressure Control During Low Temperature Operation."

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

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

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

SR 3.5.2.2

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

SR 3.5.2.3

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3 (continued)

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

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

**While lower levels of gas are ideal, due to the robust design of the ECCS system, it can still perform its design functions despite gas volumes below a specifically defined value being present in the system. An evaluation was performed on the effects of the gas on system performance which included transient effects on piping, components, and supports. The gas was also evaluated for the potential to delay the ECCS flow delivery to ensure the analyzed delivery times remained valid. An allowable gas volume for ECCS piping was calculated based on this analysis. (Ref. 8)

SR 3.5.2.4

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

SR 3.5.2.5 and 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability

BASES (continued)

(and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.7

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

SR 3.5.2.8

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

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling System."
2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plant."
3. Watts Bar FSAR, Section 6.3, "Emergency Core Cooling System."
4. FSAR Bar FSAR, Section 15.0, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987.
7. WBN License Amendment Request WBN-TS-03-11 dated April 8, 2003.
8. NEI 09-10, Revision 1a-A " Guidelines for Effective Prevention and Management of System Gas Accumulation," dated April, 2013.

BASES

BACKGROUND (continued)

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters are included to reduce the relative humidity of the airstream on systems that operate in high humidity. Operation with the heaters on for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that heater failure, blockage, fan or motor failure, or excessive vibration can be detected for corrective actions. 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.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

most repairs. If the EGTS 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 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 3.6.9.1

Operating each EGTS train for ≥ 15 minutes with heaters on ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls, the two train redundancy available.

SR 3.6.9.2

This SR verifies that the required EGTS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP-Technical Specification Section 5.7.2.14). The EGTS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). 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. It should be noted that for the EGTS, the VFTP pressure drop value across the entire filtration unit does not account for instrument error (Ref. 5).

(continued)

BASES

BACKGROUND (continued)

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the lowest MSSV setpoint (plus 3% tolerance plus 7 psi for accumulation and pressure drop between the SG and MSSV). Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The AFW System actuates automatically on steam generator water level - low-low by the ESFAS (LCO 3.3.2). The motor driven pumps start on a two-out-of-three low-low level signal in any steam generator and the turbine driven pump starts on a two-out-of-three low-low level signal in any two steam generators. The system also actuates on loss of offsite power, safety injection, and trip of both turbine-driven MFW pumps.

The AFW System is discussed in the FSAR, Section 10.4.9 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3% tolerance plus 7 psi for accumulation and pressure drop between the SG and MSSV.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

a. Feedwater Line Break (FWLB); and

b. Loss of MFW.

(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, such as from a loss of both ventilation trains or from an inoperable ABSCE boundary, could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body to the main control room occupants in the event of a large radioactive release.

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.

The LCO is modified by a Note allowing the ABSCE boundary to be opened intermittently under administrative controls that ensure the ABSCE can be closed consistent with the safety analysis. 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 are proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for auxiliary building isolation is indicated. The ABSCE boundary must be able to be restored within four minutes (including the time for restoration of the ABSCE boundary and drawdown) in accordance with UFSAR Section 15.5.3.

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, B.2 and B.3

If the ABSCE boundary is inoperable, the ABGTS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ABSCE boundary within seven days. During the period that the ABSCE boundary is inoperable, action must be initiated to implement mitigating actions consistent with the intent, as applicable, of GDC 19, 60, 61, 63, 64 and 10 CFR Part 100 (Ref. 7) to protect plant personnel from potential hazards such as radioactive contamination, temperature and relative humidity, and physical security. Actions must be taken within 24 hours to verify that, in the event of a DBA, main control room occupant radiological exposures will not exceed 10 CFR 50 Appendix A GDC 19 limits. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable ABSCE boundary) should be preplanned to address these concerns for intentional and unintentional entry into the condition. 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 seven-day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of main control room occupants within analyzed limits (Ref. 11) while limiting the probability that main control room 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 seven-day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the ABSCE boundary.

C.1 and C.2

When Required Actions A.1 or Required Actions B.1, B.2, and B.3 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable for reasons other than an inoperable ABSCE boundary (i.e., Condition B), 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.

Operation with the heaters on for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that heater failure, blockage, fan or motor failure, or excessive vibration can be detected for corrective action. 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)

BASES

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. TVA Calculation MDQ0000302014000618, “Offsite and Control Room Doses without the Auxiliary Building Secondary Containment Enclosure (ABSCE) during a LOCA.”

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WBN UNIT 1 TECHNICAL REQUIREMENTS MANUAL
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TECHNICAL REQUIREMENTS

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LIST OF ACRONYMS

Acronym	Title
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ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
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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
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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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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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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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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	01/23/2014	TRM which updates Technical Requirement (TR) 3.3.9, "Power Distribution Monitoring System," to reflect the Addendum to WCAP 12472-P-A.
Revision 55	01/14/2015	Provided in the attachment is TRM Revision 55 which revises TRM Table 3.8.3-1 pages 3 and 5, Motor-Operated Valves Thermal Overload Devices which are bypassed under accident conditions. This revision results in the valves requiring their Thermal Overload Bypasses to be operable.
Revision 56	04/30/2015	This revision restructures TR 3.6.2 CONDITIONS, REQUIRED ACTIONS, and COMPLETION TIME(s) to address two distinct cases of system inoperability. TRM BASES B 3.6.2 was also revised to coincide with the changes described above and to include additional detail regarding use of indirect means for performing channel checks
Revision 57	05/07/2015	This revision changes the elevation of the Mean Sea Level by submergence during floods vary from 714.5 ft to 739.2 ft in TRM Bases B 3.7.2, Flood Protection Plan.
Revision 58	05/19/2015	This revision is an administrative change in TRM Bases 3.4.5 background information.

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Revision 59	10/13/2015	This revision adds the Unit 1 and Unit 2 FCV-67-0066 and FCV-67-0067 valves to TRM Table 3.8.3-1.
Revision 60	06/01/2016	This revision is to add 2-FCV-70-153 valve to TRM Table 3.8.3-1 Sheet 4 of 5.
Revision 61	02/21/2017	Revises TRM Bases 3.6.2 "Inlet Door Position Monitoring System" actions.
Revision 62	03/31/2017	This revision deletes TRM and TRM Bases section 3.7.3, "Snubbers" via the License Amendment 111.
Revision 63	5/17/2017	Revises the obsolete analog system that was limited to monitoring 1 sensor for each RCS collection point.
Revision 64	8/22/17	Clarified ASME Code Class in the section description in Section 3.4.5, Piping System Structural Integrity.
Revision 65	4/6/18	Revised TRM Bases Section 3.6.2, to more closely match information provided in the UFSAR. The Bases as written limits credit for the lower inlet door main panel annunciator as part of the Inlet Door Position Monitoring System.
Revision 66 (Amendment 119)	10/11/18	Revises TRM Bases Section 3.3.5, "Turbine Overspeed Protection", to change the background information.

ENCLOSURE 4
WBN UNIT 1 TECHNICAL REQUIREMENTS MANUAL
CHANGED PAGES

B 3.3 INSTRUMENTATION

B 3.3.5 Turbine Overspeed Protection

BASES

BACKGROUND

The Digital Electro Hydraulic (DEH) system provides for redundant and diverse overspeed protection to isolate main steam to the turbo-generator when the rated operating speed of 1800 rpm is exceeded. The DEH overspeed trip function which is set at 1854 rpm (103 percent of rated speed) will initiate a turbine trip by closing all steam valves (throttle, governor, reheat, stop and interceptor valves). This trip function will trip (open) the emergency trip header valves, which will result in hydraulic closure of the aforementioned steam valves. With this arrangement, the DEH provides diverse trip functions to prevent the turbine speed from exceeding 120 percent of rated speed. DEH is a fault tolerant system, such that the loss of a single speed probe or module will not result in an unnecessary trip, but will not impede the system from performing its intended function.

If for some reason the DEH turbine trip does not function and the turbine speed increases to 1980 rpm (110 percent of rated speed), the Independent Overspeed Protection System (IOPS) will initiate a turbine trip by opening the emergency trip header valves which will then hydraulically close all steam valves (throttle, governor, reheat, stop, and interceptor valves) and prevent the turbine speed from exceeding 120 percent of rated speed. The until will then coast down to turning gear operation. The IOPS has separate speed probes and control modules, independent of any speed instrumentation from the DEH system. The IOPS is a fault tolerant system such that a failure of a single speed probe or module will not result in an unnecessary trip, but will not impede the system from performing its intended function.

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Inlet Door Position Monitoring System

BASES

BACKGROUND

Ninety-six limit switches monitor the position of the lower inlet doors. Two switches are mounted on the door frame for each door panel. The position and movement of the switches are such that the doors must be effectively sealed before the switches are actuated. A single annunciator window in the control room gives a common alarm signal when any door is open. Open/shut indication is also provided at the lower inlet door position display panel located in the Main Control Room. For door monitoring purposes, the ice condenser is divided into six zones, each containing four inlet door assemblies, or a total of eight door panels. The limit switches on the doors in any single zone are wired to a single light on the inlet door position display panel such that a closed light indicates that all the doors in that zone are shut and an open light indicates that one or more doors in that zone are open (Ref. 1). Additional information on the design of the lower inlet door monitoring instrumentation is provided in UFSAR Section 6.7.15.

Monitoring of inlet door position is necessary because the inlet doors form the barrier to air flow through the inlet ports of the ice condenser for normal unit operation. Failure of the Inlet Door Position Monitoring System requires an alternate OPERABLE monitoring system to be used to ensure that the ice condenser is not degraded.

APPLICABLE SAFETY ANALYSES

Proper operation of the inlet doors is necessary to mitigate the consequences of a loss of coolant accident or a main steam line break inside containment. The Inlet Door Position Monitoring System, however, is not required for proper operation of the inlet doors, nor is it considered OPERABLE as an initial condition for a DBA. Hence, the Inlet Door Position Monitoring System is not a consideration in the analyses of DBAs. Based on the PRA Summary Report in Reference 2, the Inlet Door Position Monitoring System has not been identified as a significant risk contributor.

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ARFS	Air Return Fan System
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ARV	Atmospheric Relief Valve
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EFPD	Effective Full-Power Days
EGTS	Emergency Gas Treatment System
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LIST OF ACRONYMS

ACRONYM	TITLE
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ESFAS	Engineered Safety Features Actuation System
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RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System

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LIST OF ACRONYMS

ACRONYM	TITLE
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

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xxi	13	B 3.1-6	0
xxii	18	B 3.1-7	0
B 2.0-1	0	B 3.1-8	0
B 2.0-2	0	B 3.1-9	0
B 2.0-3	0	B 3.1-10	0
B 2.0-4	0	B 3.1-11	0
B 2.0-5	0	B 3.1-12	0
B 2.0-6	0	B 3.1-13	0
B 2.0-7	0	B 3.1-14	0
B 2.0-8	0	B 3.1-15	0
B 2.0-9	0	B 3.1-16	0
B 2.0-10	0	B 3.1-17	0
B 3.0-1	7	B 3.1-18	0
B 3.0-2	0	B 3.1-19	0
B 3.0-3	0	B 3.1-20	0
B 3.0-4	0	B 3.1-21	0
B 3.0-5	0	B 3.1-22	0

LIST OF EFFECTIVE PAGES (continued)

PAGE NUMBER	AMENDMENT NUMBER	PAGE NUMBER	AMENDMENT NUMBER
B 3.1-23	0	B 3.1-61	0
B 3.1-24	0	B 3.1-62	0
B 3.1-25	16	B 3.1-63	0
B 3.1-26	0	B 3.1-64	0
B 3.1-27	0	B 3.1-65	0
B 3.1-28	16	B 3.1-66	0
B 3.1-29	16	B 3.1-67	0
B 3.1-30	16	B 3.1-68	0
B 3.1-31	16	B 3.1-69	0
B 3.1-32	16	B 3.1-70	0
B 3.1-33	16	B 3.2-1	0
B 3.1-34	0	B 3.2-2	0
B 3.1-35	0	B 3.2-3	0
B 3.1-36	16	B 3.2-4	0
B 3.1-37	16	B 3.2-5	0
B 3.1-38	16	B 3.2-6	0
B 3.1-39	16	B 3.2-7	0
B 3.1-40	16	B 3.2-8	0
B 3.1-41	16	B 3.2-9	0
B 3.1-42	16	B 3.2-10	0
B 3.1-43	16	B 3.2-11	0
B 3.1-44	16	B 3.2-12	0
B 3.1-45	16	B 3.2-13	0
B 3.1-46	0	B 3.2-14	0
B 3.1-47	16	B 3.2-15	0
B 3.1-48	16	B 3.2-16	0
B 3.1-49	16	B 3.2-17	0
B 3.1-50	16	B 3.2-18	0
B 3.1-51	16	B 3.2-19	0
B 3.1-52	16	B 3.2-20	0
B 3.1-53	16	B 3.2-21	0
B 3.1-54	16	B 3.2-22	0
B 3.1-55	0	B 3.2-23	0
B 3.1-56	16	B 3.2-24	0
B 3.1-57	0	B 3.2-25	0
B 3.1-58	0	B 3.2-26	0
B 3.1-59	0	B 3.2-27	0
B 3.1-60	0	B 3.2-28	0

LIST OF EFFECTIVE PAGES (continued)

PAGE NUMBER	AMENDMENT NUMBER	PAGE NUMBER	AMENDMENT NUMBER
B 3.2-29	0	B 3.3-36	0
B 3.2-30	17	B 3.3-37	0
B 3.2-31	0	B 3.3-38	17
B 3.3-1	0	B 3.3-39	17
B 3.3-2	0	B 3.3-40	0
B 3.3-3	0	B 3.3-41	0
B 3.3-4	0	B 3.3-42	0
B 3.3-5	0	B 3.3-43	0
B 3.3-6	0	B 3.3-44	0
B 3.3-7	0	B 3.3-45	0
B 3.3-8	0	B 3.3-46	0
B 3.3-9	0	B 3.3-47	0
B 3.3-10	0	B 3.3-48	0
B 3.3-11	0	B 3.3-49	0
B 3.3-12	0	B 3.3-50	0
B 3.3-13	18	B 3.3-51	0
B 3.3-14	0	B 3.3-52	0
B 3.3-15	18	B 3.3-53	0
B 3.3-16	0	B 3.3-54	0
B 3.3-17	0	B 3.3-55	0
B 3.3-18	0	B 3.3-56	0
B 3.3-19	0	B 3.3-57	0
B 3.3-20	0	B 3.3-58	0
B 3.3-21	0	B 3.3-59	0
B 3.3-22	0	B 3.3-60	0
B 3.3-23	0	B 3.3-61	0
B 3.3-24	0	B 3.3-62	0
B 3.3-25	0	B 3.3-63	0
B 3.3-26	0	B 3.3-64	0
B 3.3-27	0	B 3.3-65	0
B 3.3-28	0	B 3.3-66	0
B 3.3-29	0	B 3.3-67	0
B 3.3-30	0	B 3.3-68	0
B 3.3-31	0	B 3.3-69	0
B 3.3-32	0	B 3.3-70	0
B 3.3-33	0	B 3.3-71	0
B 3.3-34	0	B 3.3-72	0
B 3.3-35	0	B 3.3-73	0

LIST OF EFFECTIVE PAGES (continued)

PAGE NUMBER	AMENDMENT NUMBER	PAGE NUMBER	AMENDMENT NUMBER
B 3.3-74	0	B 3.3-112	0
B 3.3-75	0	B 3.3-113	0
B 3.3-76	0	B 3.3-114	0
B 3.3-77	0	B 3.3-115	0
B 3.3-78	0	B 3.3-116	0
B 3.3-79	0	B 3.3-117	0
B 3.3-80	0	B 3.3-118	0
B 3.3-81	0	B 3.3-119	0
B 3.3-82	0	B 3.3-120	0
B 3.3-83	0	B 3.3-121	0
B 3.3-84	0	B 3.3-122	0
B 3.3-85	0	B 3.3-123	0
B 3.3-86	0	B 3.3-124	0
B 3.3-87	0	B 3.3-125	0
B 3.3-88	0	B 3.3-126	0
B 3.3-89	0	B 3.3-127	0
B 3.3-90	0	B 3.3-128	0
B 3.3-91	0	B 3.3-129	0
B 3.3-92	0	B 3.3-130	0
B 3.3-93	0	B 3.3-131	0
B 3.3-94	0	B 3.3-132	0
B 3.3-95	0	B 3.3-133	0
B 3.3-96	0	B 3.3-134	0
B 3.3-97	0	B 3.3-135	0
B 3.3-98	0	B 3.3-136	0
B 3.3-99	0	B 3.3-137	0
B 3.3-100	0	B 3.3-138	0
B 3.3-101	0	B 3.3-139	0
B 3.3-102	0	B 3.3-140	0
B 3.3-103	0	B 3.3-141	0
B 3.3-104	0	B 3.3-142	0
B 3.3-105	0	B 3.3-143	0
B 3.3-106	0	B 3.3-144	0
B 3.3-107	0	B 3.3-145	0
B 3.3-108	0	B 3.3-146	0
B 3.3-109	0	B 3.3-147	0
B 3.3-110	0	B 3.3-148	0
B 3.3-111	0		

LIST OF EFFECTIVE PAGES (continued)

PAGE NUMBER	AMENDMENT NUMBER
B 3.3-149	0
B 3.3-150	0
B 3.3-151	0
B 3.3-152	0
B 3.3-153	0
B 3.3-154	0
B 3.3-155	0
B 3.3-156	0
B 3.3-157	0
B 3.3-158	0
B 3.3-159	0
B 3.3-160	0
B 3.3-161	0
B 3.3-162	0
B 3.3-163	0
B 3.3-164	0
B 3.3-165	0
B 3.3-166	0
B 3.3-167	0
B 3.3-168	0
B 3.4-1	0
B 3.4-2	0
B 3.4-3	0
B 3.4-4	0
B 3.4-5	0
B 3.4-6	0
B 3.4-7	0
B 3.4-8	0
B 3.4-9	0
B 3.4-10	0
B 3.4-11	0
B 3.4-12	0
B 3.4-13	0
B 3.4-14	0
B 3.4-15	0
B 3.4-16	0
B 3.4-17	0
B 3.4-18	0

PAGE NUMBER	AMENDMENT NUMBER
B 3.4-19	0
B 3.4-20	0
B 3.4-21	0
B 3.4-22	0
B 3.4-23	0
B 3.4-24	0
B 3.4-25	0
B 3.4-26	0
B 3.4-27	0
B 3.4-28	0
B 3.4-29	0
B 3.4-30	8
B 3.4-31	0
B 3.4-32	0
B 3.4-33	0
B 3.4-34	0
B 3.4-35	0
B 3.4-36	0
B 3.4-37	0
B 3.4-38	0
B 3.4-39	0
B 3.4-40	0
B 3.4-41	0
B 3.4-42	0
B 3.4-43	0
B 3.4-44	0
B 3.4-45	0
B 3.4-46	0
B 3.4-47	0
B 3.4-48	0
B 3.4-49	0
B 3.4-50	0
B 3.4-51	0
B 3.4-52	0
B 3.4-53	0
B 3.4-54	0
B 3.4-55	0

LIST OF EFFECTIVE PAGES (continued)

PAGE NUMBER	AMENDMENT NUMBER	PAGE NUMBER	AMENDMENT NUMBER
B 3.6-3	0	B 3.6-41	0
B 3.6-4	0	B 3.6-42	0
B 3.6-5	0	B 3.6-43	0
B 3.6-6	0	B 3.6-44	0
B 3.6-7	0	B 3.6-45	0
B 3.6-8	0	B 3.6-46	0
B 3.6-9	0	B 3.6-47	0
B 3.6-10	0	B 3.6-48	0
B 3.6-11	0	B 3.6-49	12
B 3.6-12	0	B 3.6-50	0
B 3.6-13	0	B 3.6-51	12
B 3.6-14	0	B 3.5-52	0
B 3.6-15	0	B 3.6-53	0
B 3.6-16	0	B 3.6-54	0
B 3.6-17	0	B 3.6-55	0
B 3.6-18	0	B 3.6-56	0
B 3.6-19	0	B 3.6-57	0
B 3.6-20	0	B 3.6-58	0
B 3.6-21	0	B 3.6-59	11
B 3.6-22	0	B 3.6-60	0
B 3.6-23	0	B 3.6-61	0
B 3.6-24	0	B 3.6-62	0
B 3.6-25	0	B 3.6-63	0
B 3.6-26	0	B 3.6-64	11
B 3.6-27	15	B 3.6-65	0
B 3.6-28	0	B 3.6-66	0
B 3.6-29	0	B 3.6-67	0
B 3.6-30	0	B 3.6-68	0
B 3.6-31	0	B 3.6-69	0
B 3.6-32	0	B 3.6-70	0
B 3.6-33	0	B 3.6-71	0
B 3.6-34	0	B 3.6-72	0
B 3.6-35	0	B 3.6-73	0
B 3.6-36	15	B 3.6-74	0
B 3.6-37	0	B 3.6-75	0
B 3.6-38	0	B 3.6-76	0
B 3.6-39	0	B 3.6-77	0
B 3.6-40	0		

LIST OF EFFECTIVE PAGES (continued)

PAGE NUMBER	AMENDMENT NUMBER	PAGE NUMBER	AMENDMENT NUMBER
B 3.6-78	0	B 3.7-25	0
B 3.6-79	0	B 3.7-26	0
B 3.6-80	0	B 3.7-27	0
B 3.6-81	0	B 3.7-28	0
B 3.6-82	0	B 3.7-29	0
B 3.6-83	0	B 3.7-30	0
B 3.6-84	0	B 3.7-31	0
B 3.6-85	0	B 3.7-32	0
B 3.6-86	0	B 3.7-33	0
B 3.6-87	4	B 3.7-34	0
B 3.6-88	4	B 3.7-35	0
B 3.6-89	4	B 3.7-36	6
B 3.6-90	0	B 3.7-37	0
B3.6-91	4	B 3.7-38	0
B 3.7-1	0	B 3.7-39	0
B 3.7-2	0	B 3.7-40	0
B 3.7-3	0	B 3.7-41	0
B 3.7-4	0	B 3.7-42	0
B 3.7-5	0	B 3.7-43	0
B 3.7-6	0	B 3.7-44	0
B 3.7-7	0	B 3.7-45	0
B 3.7-8	0	B 3.7-46	0
B 3.7-9	0	B 3.7-47	0
B 3.7-10	0	B 3.7-48	0
B 3.7-11	0	B 3.7-49	0
B 3.7-12	0	B 3.7-50	0
B 3.7-13	0	B 3.7-51	0
B 3.7-14	0	B 3.7-52	0
B 3.7-15	0	B 3.7-53	0
B 3.7-16	0	B 3.7-54	0
B 3.7-17	0	B 3.7-55	9
B 3.7-18	0	B 3.7-56	9
B 3.7-19	0	B 3.7-57	0
B 3.7-20	0	B 3.7-58	0
B 3.7-21	0	B 3.7-59	0
B 3.7-22	0	B 3.7-60	0
B 3.7-23	0	B 3.7-61	0
B 3.7-24	0		

LIST OF EFFECTIVE PAGES (continued)

PAGE NUMBER	AMENDMENT NUMBER
B 3.7-62	0
B 3.7-63	0
B 3.7-64	13
B 3.7-65	13
B 3.7-66	13
B 3.7-67	0
B 3.7-67a	13
B 3.7-68	0
B 3.7-69	0
B 3.7-70	0
B 3.7-71	0
B 3.7-72	0
B 3.7-73	0
B 3.7-74	0
B 3.7-75	0
B 3.7-76	0
B 3.7-77	6
B 3.7-78	6
B 3.7-79	6
B 3.7-80	0
B 3.7-81	0
B 3.7-82	0
B 3.7-83	0
B 3.7-84	0
B 3.7-85	0
B 3.7-86	0
B 3.7-87	0
B 3.7-88	0
B 3.8-1	0
B 3.8-2	5
B 3.8-3	0
B 3.8-4	14
B 3.8-5	0
B 3.8-6	0
B 3.8-7	0
B 3.8-8	0
B 3.8-9	5
B 3.8-10	5

PAGE NUMBER	AMENDMENT NUMBER
B 3.8-10a	5
B 3.8-11	5
B 3.8-12	5
B 3.8-12a	5
B 3.8-12b	5
B 3.8-13	5
B 3.8-14	5
B 3.8-15	5
B 3.8-16	5
B 3.8-17	0
B 3.8-18	0
B 3.8-19	0
B 3.8-20	0
B 3.8-21	0
B 3.8-22	0
B 3.8-23	0
B 3.8-24	0
B 3.8-25	0
B 3.8-26	0
B 3.8-27	0
B 3.8-28	0
B 3.8-28a	5
B 3.8-29	0
B 3.8-30	0
B 3.8-31	0
B 3.8-32	0
B 3.8-33	0
B 3.8-34	0
B 3.8-35	0
B 3.8-36	0
B 3.8-37	0
B 3.8-37a	13
B 3.8-38	0
B 3.8-39	0
B 3.8-40	0
B 3.8-41	0
B 3.8-42	0

LIST OF EFFECTIVE PAGES (continued)

PAGE NUMBER	AMENDMENT NUMBER	PAGE NUMBER	AMENDMENT NUMBER
B 3.8-43	0	B 3.8-81	0
B 3.8-44	0	B 3.8-82	0
B 3.8-45	0	B 3.8-83	0
B 3.8-46	0	B 3.8-84	0
B 3.8-47	0	B 3.8-85	0
B 3.8-48	0	B 3.8-86	0
B 3.8-49	0	B 3.8-87	0
B 3.8-50	0	B 3.8-88	0
B 3.8-51	0	B 3.8-89	0
B 3.8-52	0	B 3.8-90	0
B 3.8-53	0	B 3.8-91	0
B 3.8-54	0	B 3.8-92	0
B 3.8-55	0	B 3.8-93	0
B 3.8-56	0	B 3.8-94	1
B 3.8-57	0	B 3.8-95	0
B 3.8-58	0	B 3.8-96	0
B 3.8-59	0	B 3.8-97	0
B 3.8-60	0	B 3.8-98	0
B 3.8-61	0	B 3.9-1	0
B 3.8-62	0	B 3.9-2	0
B 3.8-63	0	B 3.9-3	0
B 3.8-64	0	B 3.9-4	0
B 3.8-65	0	B 3.9-5	0
B 3.8-66	0	B 3.9-6	0
B 3.8-67	0	B 3.9-7	0
B 3.8-68	0	B 3.9-8	0
B 3.8-69	0	B 3.9-10	0
B 3.8-70	0	B 3.9-11	0
B 3.8-71	0	B 3.9-12	0
B 3.8-72	0	B 3.9-13	0
B 3.8-73	0	B 3.9-14	0
B 3.8-74	0	B 3.9-15	0
B 3.8-75	0	B 3.9-16	0
B 3.8-76	0	B 3.9-17	0
B 3.8-77	0	B 3.9-18	0
B 3.8-78	0	B 3.9-19	0
B 3.8-79	0	B 3.9-20	0
B 3.8-80	0	B 3.9-21	0

LIST OF EFFECTIVE PAGES (continued)

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
NPF-20	10-22-15	Low Power Operating License
Revision 1	2-12-16	TS Bases Table B 3.8.9-1, "AC and DC Electrical Power Distribution Systems"
Revision 2	3-18-16	Revise TS Bases B3.3.7, "Component Cooling System (CCS)," regarding the 1B and 2B surge tank sections.
Revision 3	7-11-16	Revise TS Bases B3.6.4, "Containment Pressure," and B3.6.6, "Containment Spray System" regarding the maximum peak containment pressure from a LOCA of 11.73 psig.
Revision 4	8-19-16	Revise TS Bases B3.6.15, "Shield Building," to clarify the use of the Condition B note.
Revision 5	1-17-17	Revises TS Bases B 3.8.1 "AC-Sources"
Revision 6	2-24-17	Revises TS Bases B 3.7.7, "Component Cooling System (CCS)," and B 3.7.16, "Component Cooling System (CCS) - Shutdown".
Revision 7	3-13-17	Adds TS Bases B 3.0.8 for Inoperability of Snubbers.
Revision 8	4-7-17	Revises TS Bases B 3.4.6.3 to correct the steam generator minimum narrow range level.
Revision 9	4-25-17	Revises TS Bases B3.7-10 CREVS.
Revision 10	7-14-17	Revises TS Bases SR B3.0.2 for a one-time extension of the Alternating Current Sources.
Revision 11, Amendment 14	9-29-17	Revises TS Bases B3.6.11 to change the ice mass weight.
Revision 12, Amendment 15	11-2-17	Revises TS Bases to adopt the TSTF-522 to revise ventilation system surveillance requirements to operate for 10 hours per month.
Revision 13, Amendment 16	11-2-17	Revises TS Bases B3.7-12 to provide action when both trains of ABGTS are inoperable. Also, B3.8-37a correction of unit error.

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 14	11-9-17	Revises TS Bases B 3.8.1 AC Sources - Operating LCO to correct a typo 1.a.
Revision 15	12-13-17	Revises TS Bases B3.6.4 and B 3.6.6 to change the calculated peak pressure.
Revision 16, Amendment 20	08-20-18	Revises TS Bases B3.1.5, B3.1.6, B3.1.7, and B3.1.8 which adopts the TSTF-547, Clarification of Rod position requirements.
Revision 17, Amendment 21	09-21-18	Revises TS Bases 3.2.4 and Bases 3.3.1 related to the reactor trip system instrumentation.
Revision 18	02-13-19	Revises TS Bases 3.3.1 related to the reactor trip system instrumentation.

ENCLOSURE 6
WBN UNIT 2 TECHNICAL SPECIFICATION BASES
CHANGED PAGES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy and Capability," (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause a control or shutdown rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each). A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. Except for Shutdown Banks C and D,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly using incore power distribution measurements. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

BASES (continued)

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - T_{avg} > 200°F," for SDM in MODES 3 and 4, LCO 3.1.2, "Shutdown Margin (SDM) - $T_{avg} \leq$ 200°F" for SDM in MODE 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating boration to restore SDM.

In this situation, SDM verification must include the worth of the untrippable rod, as well as a rod of maximum worth.

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

ACTIONS (continued)

B.1.1 and B.1.2

When a rod becomes misaligned, it can usually be moved and is still trippable.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7, "Control Bank Insertion Limits."

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

(continued)

BASES

ACTIONS (continued)

B.2, B.3, B.4, and B.5

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 6). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain an incore power distribution measurement and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in UFSAR Chapter 15 (Ref. 3) that may be adversely affected will be evaluated to ensure that the analyses remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

(continued)

ACTIONS (continued)

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable.

To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the position of individual rods is within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

The SR is modified by a NOTE that permits it to not be performed for rods associated with an inoperable demand position indicator or an inoperable rod position indicator. The alignment limit is based on the demand position indicator which is not available if the indicator is inoperable. LCO 3.1.8, “Rod Position Indication,” provides Actions to verify the rods are in alignment when one or more rod position indicators are inoperable.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.1 (continued)

The Surveillance is modified by a NOTE which states that the SR is not required to be performed until 1 hour after associated rod motion. Control rod temperature affects the accuracy of the rod position indication system. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows control rod temperature to stabilize following rod movement in order to ensure the indicated rod position is accurate.

SR 3.1.5.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.5.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.5.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.5.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality after each reactor vessel head removal ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 551°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

BASES

BACKGROUND (continued)

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are then left in this position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.7, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - $T_{avg} > 200°F$," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - $T_{avg} \leq 200°F$") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There be no violations of:

   1. Specified acceptable fuel design limits, or

   2. RCS pressure boundary integrity; and

b. The core remains subcritical after accident transients other than a main steam line break (MSLB).

(continued)

APPLICABLE SAFETY ANALYSES (continued)	As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3). The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown bank insertion limits are defined in the COLR. The LCO is modified by a Note indicating the LCO requirement is not applicable to shutdown banks being inserted while performing SR 3.1.5.2. This SR verifies the freedom of the rods to move, and may require the shutdown bank to move below the LCO limits, which would normally violate the LCO. This Note applies to each shutdown bank as it is moved below the insertion limit to perform the SR. This Note is not applicable should a malfunction stop performance of the SR.
APPLICABILITY	The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. Refer to LCO 3.1.1 and LCO 3.1.2 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

BASES

ACTIONS

A.1, A.2.1, A.2.2 and A.3

If one shutdown bank is inserted less than or equal to 10 steps below the insertion limit, 24 hours is allowed to restore the shutdown bank to within the limit. This is necessary because the available SDM may be reduced with a shutdown bank not within its insertion limit. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If a shutdown bank is not within its insertion limit, SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

While the shutdown bank is outside the insertion limit, all control banks must be within their insertion limits to ensure sufficient shutdown margin is available. The 24 hour Completion Time is sufficient to repair most rod control failures that would prevent movement of a shutdown bank.

B.1.1, B.1.2 and B.2

When one or more shutdown banks is not within insertion limits for reasons other than Condition A, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (See LCO 3.1.1.). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

C.1

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

The Surveillance is modified by a Note which states that the SR is not required to be performed for shutdown banks until 1 hour after motion of rods in those banks. Rod temperature affects the accuracy of the rod position indication system. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows rod temperature to stabilize following rod movement in order to ensure the indicated position is accurate.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12-hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," General Design Criterion 26, "Reactivity Control System Redundancy and Capability," and General Design Criterion 28, "Reactivity Limits."

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

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," and GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each). A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. Except for Shutdown Banks C and D, a bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and four shutdown banks. See LCO 3.1.5, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. An example is provided for information only in Figure B 3.1.7-1. The control banks are required to be at or above the insertion limit lines.

Figure B 3.1.7-1 also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal, as an example may be at 116 steps. Therefore, in this example, control bank C overlaps control bank D from 116 steps to the fully withdrawn position for control bank C. The fully withdrawn position and predetermined overlap positions are defined in the COLR.

(continued)

BASES

BACKGROUND (continued)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Rod Control System, but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.5, "Rod Group Alignment Limits," LCO 3.1.6, "Shutdown Bank Insertion Limits," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

   1. Specified acceptable fuel design limits, or

   2. Reactor Coolant System pressure boundary integrity; and

b. The core remains subcritical after accident transients other than a main steam line break (MSLB).

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3 through 13).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 5, 6, 8 and 11). Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths. The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 3 through 13). The insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analysis.
LCO	The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion. The LCO is modified by a Note indicating the LCO requirement is not applicable to control banks being inserted while performing SR 3.1.5.2. This SR verifies the freedom of the rods to move, and may require the control bank to move below the LCO limits, which would normally violate the LCO. This Note applies to each control bank as it is moved below the insertion limit to perform the SR. This Note is not applicable should a malfunction stop performance of the SR.
APPLICABILITY	The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

(continued)

ACTIONS

A.1, A.2.1, A.2.2, and A.3

If Control Bank A, B, or C is inserted less than or equal to 10 steps below the insertion, sequence, or overlap limits, 24 hours is allowed to restore the control bank to within the limits. Verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (See LCO 3.1.1). If a control bank is not within its insertion limit, SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

While the control bank is outside the insertion, sequence, or overlap limits, all shutdown banks must be within their insertion limits to ensure sufficient shutdown margin is available and that power distribution is controlled. The 24 hour Completion Time is sufficient to repair most rod control failures that would prevent movement of a shutdown bank.

Condition A is limited to Control Banks A, B, or C. The allowance is not required for Control Bank D because the full power bank insertion limit can be met during performance of the SR 3.1.5.2 control rod freedom of movement (trippability) testing.

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

When the control banks are outside the acceptable insertion limits for reasons other than Condition A, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) - T_{avg} > 200°F") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration for reasons other than Condition A, they must be restored to meet the limits.

(continued)

ACTIONS

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

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

D.1

If the Required Actions cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with k_{eff} < 1.0, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.7.2

With an OPERABLE bank insertion limit monitor, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance is modified by a Note stating that the SR is not required to be performed for control banks until 1 hour after motion of rods in those banks. Control rod temperature affects the accuracy of the rod position indication system. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows control rod temperature to stabilize following rod movement in order to ensure the indicated rod position is accurate.

SR 3.1.7.3

When control banks are maintained within their insertion limits as checked by SR 3.1.7.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR.

The Surveillance is modified by a Note stating that the SR is not required to be performed for control banks until 1 hour after motion of rods in those banks. Control rod temperature affects the accuracy of the rod position indication system. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows control rod temperature to stabilize following rod movement in order to ensure the indicated rod position is accurate.

A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.7.2

(continued)

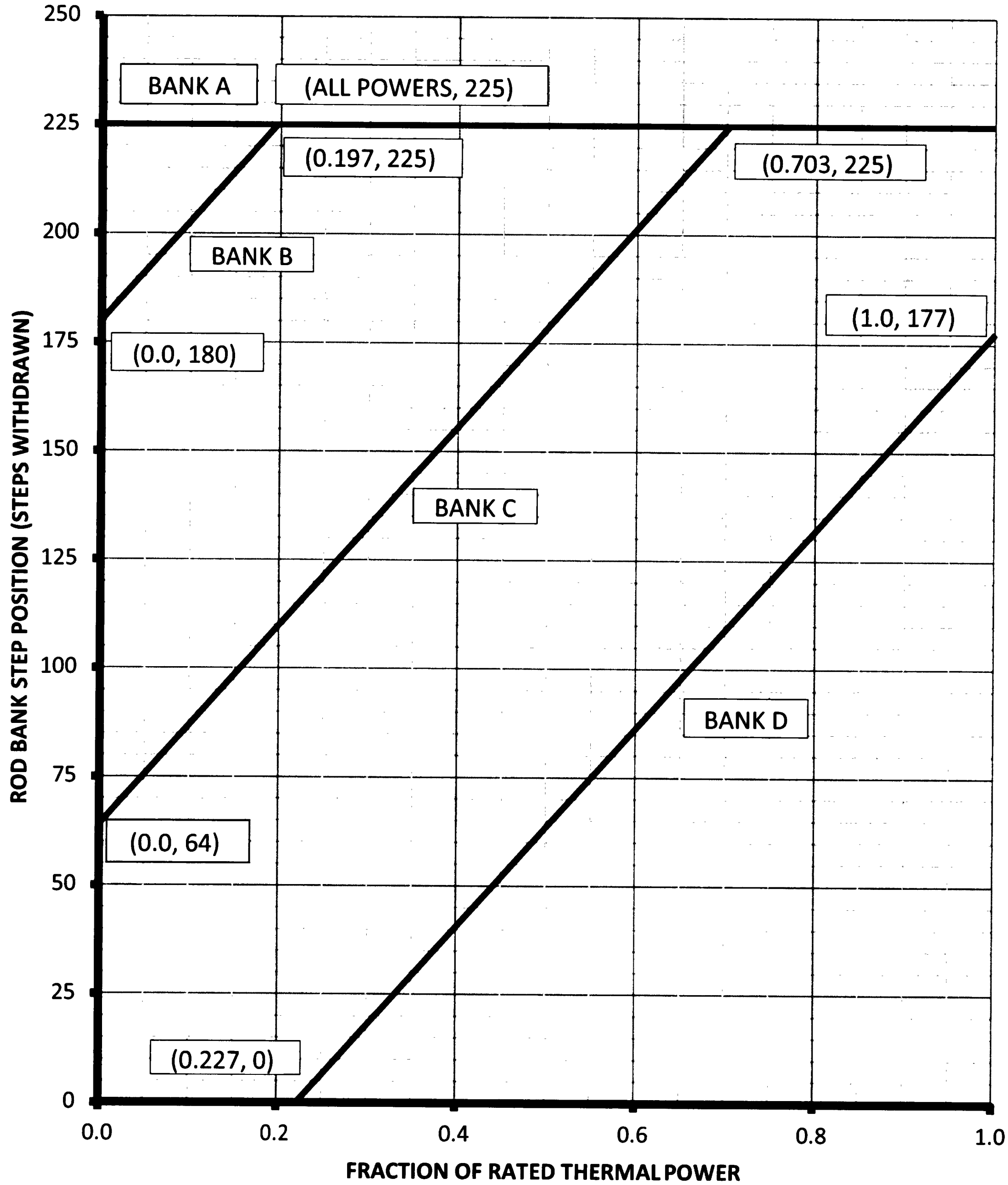

Figure B 3.1.7-1
CONTROL BANK INSERTION
vs. RTP

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.8 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control (Shutdown Banks C and D have only one group each).

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Rod Position Indication (RPI) System.

(continued)

BASES

BACKGROUND (continued)

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The RPI System provides an accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of alternating primary and secondary coils spaced along a hollow tube. The normal indication accuracy of the RPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps (± 7.5 inches). With an indicated deviation of 12 steps between the group step counter and RPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

The Power Distribution Monitoring System (PDMS) as controlled by Technical Requirements Manual Section 3.3.9 develops a detailed three dimensional power distribution via its nodal code coupled with updates from plant instrumentation, including the fixed incore detectors. The monitored power distribution is compared to the reference power distribution corresponding to all control rods properly aligned. Agreement between the two power distributions can be used to indirectly verify the control rod is aligned.

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2 through 12), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.5, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

BASES (continued)

LCO

LCO 3.1.8 specifies that the RPI System and the Bank Demand Position Indication System be OPERABLE for all control rods. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO (when required) and the following:

a. The RPI System indicates within 12 steps of the group step counter demand position when LCO 3.1.5, "Rod Group Alignment Limits;" met.

b. For the RPI System there are no failed coils; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the RPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the RPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.5, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

The LCO is modified by a NOTE stating that the RPI system is not required to be OEPRABLE for 1 hour following movement of the associated rods. Control and shutdown rod temperature affects the accuracy of the RPI System. Due to changes in the magnetic permeability of the drive shaft as a function of temperature, the indicated position is expected to change with time as the drive shaft temperature changes. The one hour period allows temperature to stabilize following rod movement in order to ensure the indicated position is accurate.

(continued)

BASES (continued)

APPLICABILITY

The requirements on the RPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1 and A.2

When one RPI channel per group in one or more groups fails, the position of the rod can still be determined indirectly by use of incore power distribution measurement information. Incore power distribution measurement information is obtained from an OPERABLE Power Distribution Monitoring System (PDMS) (Ref. 15). The Required Action may also be satisfied by ensuring at least once per 8 hours that F_Q satisfies LCO 3.2.1, $F^N_{\Delta H}$ satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the COLR, provided the non-indicating rods have not been moved. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of rod position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

Required Action A.1 requires verification of a rod with an inoperable RPI once per 8 hours. Required Action A.2 provides an alternative. Required Action A.2 requires verification of rod position using incore power distribution measurement information every 31 EFPD, which coincides with the normal measurements to verify core power distribution.

Required Action A.2 includes six distinct requirements for verification of the position of rods associated with an inoperable RPI using incore power distribution measurement information:

a. Initial verification within 8 hours of the inoperability of the RPI;

(continued)

BASES (continued)

ACTIONS

A.1 and A.2 (continued)

b. Re-verification once every 31 Effective Full Power Days (EFPD) thereafter;

c. Verification within 8 hours after discovery of each unintended rod movement. An unintended rod movement is defined as the release of the rod's stationary gripper when no action was demanded either manually or automatically from the rod control system, or a rod motion in a direction other than the direction demanded by the rod control system. Verifying that no unintended rod movement has occurred is performed by monitoring the rod control system stationary gripper coil current for indications of rod movement;

d. Verification within 8 hours if the rod with an inoperable RPI is intentionally moved greater than 12 steps;

e. Verification prior to exceeding 50% RTP if power is reduced below 50% RTP; and

f. Verification within 8 hours of reaching 100% RTP if power is reduced to less than 100% power RTP.

Should the rod with the inoperable RPI be moved more than 12 steps, or if reactor power is changed, the position of the rod with the inoperable RPI must be verified.

A.3

Reduction of THERMAL POWER to ≤ 50% RTP puts core into a condition where rod position is not significantly affecting core peaking factors (Ref. 4). The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Actions A.1 and A.2 above.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2

When more than one RPI per group in one or more groups fail, additional actions are necessary. Placing the Rod Control System in manual assures unplanned rod motion will not occur. The immediate Completion Time for placing the Rod Control System in manual reflects the urgency with which unplanned rod motion must be prevented while in this Condition.

The inoperable RPIs must be restored, such that a maximum of one RPI per group is inoperable, within 24 hours. The 24 hour Completion Time provides sufficient time to troubleshoot and restore the RPI system to operation while avoiding the plant challenges associated with the shutdown without full rod position indication.

Based on operating experience, normal power operation does not require excessive rod movement. If one or more rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

C.1 and C.2

With one or more RPI inoperable in one or more groups and the affected groups have moved greater than 24 steps in one direction since the last determination of rod position, additional actions are needed to verify the position of rods with inoperable RPI. Within 4 hours, the position of the rods with inoperable position indication must be determined using the PDMS to verify these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been verified, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

D.1.1 and D.1.2

With one or more demand position indicators per bank inoperable in one or more banks, the rod positions can be determined by the RPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

(continued)

BASES

ACTIONS (continued)

D.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits (Ref. 13). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power ≤ 50% RTP.

E.1

If the Required Actions cannot be completed within the associated Completion Time, 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. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the RPI agrees with the demand position within 12 steps ensures that the RPI is operating correctly.

This Surveillance is performed prior to reactor criticality after each removal of the reactor head, as there is the potential for unnecessary plant transients if the SR were performed with the reactor at power.

The Surveillance is modified by a NOTE which states it is not required to be met for RPIs associated with rods that do not meet LCO 3.1.5. If a rod is known to not be within 12 steps of the group demand position, ACTIONS of LCO 3.1.5 provide the appropriate Actions.

BASES (continued)

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BASES

ACTIONS (continued)

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one power range neutron flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1 if more than one input from power range neutron flux channels are inoperable.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those changes of QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states the surveillance is only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER >75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Power Range Neutron Flux - High Positive Rate (continued)

The LCO requires all four of the Power Range Neutron Flux - High Positive Rate channels to be OPERABLE.

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux - High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this MODE.

b. Power Range Neutron Flux - High Negative Rate

Deleted

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides backup protection to the Power Range Neutron Flux - Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. This Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip.

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

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. Source Range Neutron Flux (continued)

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux - Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS Source Range Neutron Flux trip Function may be manually blocked. Above the P-10 setpoint, the NIS Source Range Neutron Flux trip function is automatically blocked.

In MODE 3, 4, or 5 with the reactor shut down, the Source Range Neutron Flux trip Function must also be OPERABLE. If the CRD System is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the CRD System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide visual indication and audible alarm of reactivity changes that may occur as a result of events like a boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressurizer pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip

(continued)

ACTIONS

C.1 and C.2 (continued)

must be opened within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, these Functions are no longer required. The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE channel or train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

D.1 and D.2

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 by Required Action D.1 to place the inoperable channel in the tripped condition is justified in Reference 14.

The Required Actions have been modified by two Notes. Note 1 allows the inoperable channel to be placed in the bypassed condition for up to 12 hours while performing routine surveillance testing of other channels. With one channel inoperable, the Note also allows routine surveillance testing of another channel with the inoperable channel in bypass. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the Power Range Neutron Flux-High setpoint in accordance with other Technical Specifications. The 12 hour time limit is justified in Reference 14.

Note 2 states to perform SR 3.2.4.2 if input to QPTR from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP.

If Required Action D.1 cannot be met within the specified Completion Time, 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 6 hours for the MODE reduction as required by Required Action D.2. 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.

(continued)

BASES

ACTIONS (continued)

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)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND	The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential (-2.0 psid) with respect to the shield building annulus atmosphere in the event of inadvertent actuation of the Containment Spray System or Air Return Fans. Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.
APPLICABLE SAFETY ANALYSES	Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1). The initial pressure condition used in the containment analysis was 15.0 psia. This resulted in a maximum peak containment pressure from a LOCA of 9.36 psig. The containment analysis (Ref. 1) shows that the maximum allowable internal containment pressure, P_a (15.0 psig), bounds the calculated results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA does not exceed the maximum allowable calculated containment pressure of 15.0 psig.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment 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 9.36 psig results from the LOCA analysis, and is calculated to be less than the containment maximum allowable pressure of 15 psig. The maximum peak containment atmosphere temperature results from the SLB analysis. The calculated transient containment atmosphere temperatures are acceptable for the DBA SLB.

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 234 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 10 CFR 50.36(c)(2)(ii).

BASES

BACKGROUND (continued)

The prefilters remove large particles in the air, and the moisture separators remove entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal absorbers. Heaters are included to reduce the relative humidity of the airstream on systems that operate in high humidity. Operation with the heaters on for ≥15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that heater failure, blockage, fan or motor failure, or excessive vibration can be detected for corrective action. 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 pressure control operators (PCO). 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 -0.5 inches water gauge 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.

(continued)

BASES

ACTIONS (continued)

A.1

With one EGTS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7-day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant EGTS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

B.1 and B.2

If the EGTS 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 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 3.6.9.1

Operating each EGTS train for ≥ 15 minutes with heaters on ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31-day Frequency was developed in consideration of the known reliability of fan motors and controls, the two train redundancy available.

SR 3.6.9.2

This SR verifies that the required EGTS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP - Technical Specification Section 5.7.2.14). The EGTS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). 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).

(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. 5).

The ABGTS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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, such as from a loss of both ventilation trains or from an inoperable ABSCE boundary, could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body to the main control room occupants in the event of a large radioactive release.

The ABGTS is considered OPERABLE when the individual components necessary to control exposure in the Auxiliary 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.

The LCO is modified by a Note allowing the ABSCE boundary to be opened intermittently under administrative controls that ensure the ABSCE can be closed consistent with the safety analysis. 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 are proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for auxiliary building isolation is indicated. The ABSCE boundary must be able to be restored within four minutes (including the time for restoration of the ABSCE boundary and drawdown) in accordance with UFSAR Section 15.5.3.

(continued)

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.

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, B.2, and B.3

If the ABSCE boundary is inoperable, the ABGTS trains cannot perform their intended functions. Actions must be taken to restore an OEPRABLE ABSCE boundary within seven days. During the period that the ABSCE boundary is inoperable, action must be initiated to implement mitigating actions consistent with the intent, as applicable, of GDC 19, 60, 61, 63, 64 and 10 CFR Part 100 (Ref. 6) to protect plant personnel from potential hazards such as radioactive contamination, temperature and relative humidity, and physical security. Actions must be taken within 24 hours to verify that, in the event of a DBA, main control room occupant radiological exposures will not exceed 10 CFR 50 Appendix A GDC 19 limits. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable ABSCE boundary) should be preplanned to address these concerns for intentional and unintentional entry into the condition. 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 seven-day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of main control room occupants within analyzed limits (Ref. 9) while limiting the probability that main control room 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 seven-day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the ABSCE boundary.

(continued)

BASES (continued)

ACTIONS (continued)

C.1 and C.2

When Required Action A.1 or Required Actions B.1, B.2, and B.3 cannot be completed within the associated Completion Time, or when both ABGTS trains are inoperable for reasons other than an inoperable ABSCE boundary (i.e., Condition B), 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.

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.

Operation with the heaters on for ≥ 15 continuous minutes demonstrates OPERABILITY of the system. Periodic operation ensures that heater failure, blockage, fan or motor failure, or excessive vibration can be detected for corrective action. 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. 7). 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)

BASES

REFERENCES (continued)

7. 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.”

8. NUREG-0800, Section 6.5.1, “Standard Review Plan,” Rev. 2, “ESF Atmosphere Cleanup System,” July 1981.

9. TVA Calculation MDQ0000302014000618, “Offsite and Control Room Doses without the Auxiliary Building Secondary Containment Enclosure (ABSCE) during a LOCA.”

LCO (continued)

to power two shutdown boards in the same load group through either CSST A or CSST B and its associated Unit Boards, either directly from the CSST through the Unit Board or by automatic transfer from the Unit Station Service Transformer (USST) to the CSST. Use of CSST A or B as an offsite source requires that CSST A and B both be available and that the associated power and control feeders be in their normal positions to ensure independence. Due to independence limitations, CSST A and B cannot be credited for supply of both the offsite power sources simultaneously. The medium voltage power system starts at the low-side of the common station service transformers.

Each required offsite circuit is that combination of power sources described below that are either connected to the Class 1E AC Electrical Power Distribution System, or is available to be connected to the Class 1E AC Electrical Power Distribution System through automatic transfer at the 6.9 kV Shutdown or Unit Boards within a few seconds as required.

The following offsite power configurations meet the requirements of the LCO:

1. Normal Operation (i.e., all 6.9 kV shutdown boards aligned to their normal offsite circuit) – Two offsite circuits consisting of (a) AND (b) (no board transfers required; a loss of either circuit will not prevent the minimum safety functions from being performed);

   a. From the 161 kV Watts Bar Hydro Switchyard (Bay 13), through CSST C (winding Y) to 6.9 kV Shutdown Board 1A-A and (winding X) to 6.9kV Shutdown Board 2A-A: AND

   b. From the 161 kV Watts Bar Hydro Switchyard (Bay 4), through CSST D (winding X) to 6.9 kV Shutdown Board 1B-B and (winding Y) to 6.9 kV Shutdown Board 2B-B.

2. Alternate Operation (i.e., one or more 6.9 kV shutdown boards aligned to their alternate offsite circuit) – Two offsite circuits consisting of (a) AND (b) AND (c) (as needed) (Note: 6.9 kV shutdown board(s) aligned to normal circuit require an OPERABLE automatic transfer; a loss of either circuit will not prevent the minimum safety functions from being performed);

   a. From the 161 kV Watts Bar Hydro Switchyard (Bay 13), through CSST C (winding Y) to 6.9 kV Shutdown Board 1A-A (normal) AND/OR Shutdown Board 2B-B (alternate) and (winding X) to 6.9 kV Shutdown Board 2A-A (normal) AND/OR Shutdown Board 1B-B (alternate);

(continued)

Bases Table 3.8.1-2
TS Action or Surveillance Requirement (SR) Contingency Actions

	Contingency Actions to be Implemented	Applicable TS Action or SR	Applicable Modes
1.	Verify that the offsite power system is stable. This action will establish that the offsite power system is within single-contingency limits and will remain stable upon the loss of any single component supporting the system. If a grid stability problem exists, the planned DG outage will not be scheduled.	SR 3.8.1.14 Action B.5	1, 2 1, 2, 3, 4
2.	Verify that no adverse weather conditions are expected during the outage period. The planned DG outage will be postponed if inclement weather (such as severe thunderstorms or heavy snowfall) is projected.	SR 3.8.1.14 Action B.5	1, 2 1, 2, 3, 4
3.	Do not remove from service the ventilation systems for the 6.9 kV shutdown boardrooms, the elevation 772 transformer rooms, or the 480-volt shutdown board rooms, concurrently with the DG, or implement appropriate compensatory measures.	Action B.5	1, 2, 3, 4
4.	Do not remove the reactor trip beakers from service concurrently with planned DG outage maintenance.	Action B.5	1, 2, 3, 4
5.	Do not remove the turbine-driven auxiliary feedwater (AFW) pump from service concurrently with a Unit 2 DG outage.	Action B.5	1, 2, 3, 4
6.	Do not remove the AFW level control valves to the steam generators from service concurrently with a Unit 2 DG outage	Action B.5	1, 2, 3, 4
7.	Do not remove the opposite train residual heat remove (RHR) pump from service concurrently with a Unit 2 DG outage.	Action B.5	1, 2, 3, 4

ENCLOSURE 7
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LIST OF ACRONYMS

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ACRONYM	TITLE
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARFS	Air Return Fan System
ARO	All Rods Out
ARV	Atmospheric Relief Valve
ASME	American Society of Mechanical Engineers
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

(continued)

ACRONYM	TITLE
MTC	Moderator Temperature Coefficient
N/A	Not Applicable
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
TSR	Technical Surveillance Requirement
UHS	Ultimate Heat Sink

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B 3.4-14	0	B 3.7-21	0
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Revisions	Issued	SUBJECT
Revision 01	11/25/15	Revises TRM and TRM Bases section 3.7.3, "Snubbers".
Revision 02	05/22/16	TR Table 3.3.1-1, "Reactor Trip System Instrumentation Response Times" , to change the overtemperature and over power times.
Revision 03	06/27/16	TR Table 3.8.3-1, "Motor-Operated Valves Thermal Overload Devices which are Bypassed under Accident Conditions", add valve 2-FCV-70-133 and delete 4 obsolete valves.
Revision 04	02/21/17	Revises TRM Bases 3.6.2, "Inlet Door Position Monitoring System," Actions.
Revision 05	03/31/17	Revises TRM and TRM Bases to delete section 3.7.3 "Snubbers."
Revision 06	07/08/17	Revises TRM section 3.0, "Technical Surveillance Requirements (TSR) Applicability" and adds Table 3.0.2-1.
Revision 07	08/22/17	Revises the TR 3.4.5 Title to add ASME Class 1, 2, and 3 in the TRM and Bases. Also revised TSR Table 3.0.2-1 to add two addition TSRs.
Revision 08	03/08/18	Revises TR Table 3.8.4-1 to revise the dual fan motors which were replaced with single fan motors.
Revision 09	04/06/18	Revises TRM Bases B3.6.2 to more closely match information provided in the UFSAR. The Bases as written limits credit for the lower inlet door main panel annunciator as part of the Inlet Door Position Monitoring system.
Revision 10	04/27/18	Revises TRM Table 3.7.5-1, Item 9 to correct the unit identifier on the Mechanical Equipment Room.

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Table 3.7.5-1 (Page 1 of 2)

Area Temperature Monitoring

	AREA	NORMAL LIMIT (°F)	ABNORMAL LIMIT (°F)
1.	Aux Bldg el 772 next to 480V Sd Bd transformer 1A2-A.	≤ 104	≤ 110
2.	Aux Bldg el 772 next to 480V Sd Bd transformer 1B1-B.	≤ 104	≤ 110
3.	Aux Bldg el 772 next to 480V Sd Bd transformer 2A2-A.	≤ 104	≤ 110
4.	Aux Bldg el 772 next to 480V Sd Bd transformer 2B2-B.	≤ 104	≤ 110
5.	Aux Bldg el 772 next to 480V Rx MOV Bd 1A2-A.	≤ 83	≤ 104
6.	Aux Bldg el 772 next to 480V Rx MOV Bd 2A2-A.	≤ 83	≤ 104
7.	Aux Bldg el 772 next to 480V Rx MOV Bd 2B2-B.	≤ 83	≤ 104
8.	Aux Bldg el 772 across from spare 125V vital battery charger 1-S.	≤ 83	≤ 104
9.	Aux Bldg el 772 U2 Mech Equip Room.	≤ 91	≤ 104
10.	Aux Bldg el 757 U1 Sd Bd room behind stairs S-A3.	≤ 85	≤ 104
11.	Aux Bldg el 757 U2 Sd Bd room behind stairs S-A13.	≤ 85	≤ 104
12.	Aux Bldg el 757 U1 Refueling beside Aux boration makeup tk.	≤ 104	≤ 115
13.	Aux Bldg el 737 U2 outside supply fan room.	≤ 104	≤ 110
14.	Aux Bldg el 713 U2 across from AFW pumps.	≤ 104	≤ 110
15.	Aux Bldg el 692 U2 outside AFW pump room door.	≤ 104	≤ 110
16.	Aux Bldg el 692 U2 near boric acid concentrate filter vault.	≤ 104	≤ 110
17.	Aux Bldg el 676 next to O-L-629.	≤ 104	≤ 110
18.	North steam valve vault room U2. (at affected MSSVs)	≥ 50	≥ 50
19.	South steam valve vault room U2. (at affected MSSVs)	≥ 50	≥ 50

(continued)

Table 3.8.4-1 (Page 1 of 3)

Submerged Components With Automatic De-energization Under Accident Conditions

BOARD	COMPT	LOAD	FCTN
6.9kV SHUTDOWN BOARD 2A-A	20	2-DPL-68-341A-A	SI
	21	2-DPL-68-341F	SI
6.9kV SHUTDOWN BOARD 2B-B	20	2-DPL-68-341D-B	SI
	21	2-DPL-68-341H*	SI
480V SHUTDOWN BOARD 2A1-A	7B	2-MTR-30-83-A	CIB
	7C	2-MTR-30-74-A	CIB
480V SHUTDOWN BOARD 2B1-B	7C	2-MTR-30-92-B	CIB
	7D	2-MTR-30-75-B	CIB
480V SHUTDOWN BOARD 2A2-A	7A	2-MTR-30-88-A	CIB
	7D	2-MTR-30-77-A	CIB
480V SHUTDOWN BOARD 2B2-B	7B	2-MTR-30-80-B	CIB
	7D	2-MTR-30-78-B	CIB
480V REACTOR MOV BOARD 2A1-A	16A	2-MTR-31-265	CIA
	17E	2-PO-213-A1/(1-5)	SI
	18F2	2-PO-213-A1/(6-10)	SI
480V REACTOR MOV BOARD 2B1-B	16A	2-MTR-31-266	CIA
	16E	2-PO-213-B1/(1-5)	SI
	17E	2-PO-213-B1/(6-10)	SI

(continued)

Table 3.8.4-1 (Page 3 of 3)

Submerged Components With Automatic De-energization Under Accident Conditions

BOARD	COMPT	LOAD	FCTN
125VDC VITAL BATTERY BOARD IV	A20	2-FCV-43-2-B	CIA
	A21	2-FCV-43-11-B	CIA
	A22	2-FCV-43-22-B	CIA
	A23	2-FCV-43-34-B	CIA
	A24	2-FCV-77-16-B	CIA
	A43	2-FCV-77-127-B	CIA
	A44	2-FCV-77-9-B	CIA
	A45	2-FCV-77-18-B	CIA
	B26	2-FCV-30-8/50-B	CVI
	B27	2-FCV-90-108-B	CVI
	B28	2-FCV-90-110-B	CVI
	B32	2-FCV-30-15/57-B	CVI
	B33	2-FCV-90-114-B	CVI
	B34	2-FCV-30-58-B	CVI
	B36	2-FCV-90-116-B	CVI
	C5	2-FCV-31-327-B	CIA
	C6	2-FCV-30-329-B	CIA
	C26	2-FCV-61-122-B	CIA
	C34	2-FCV-90-109-B	CVI
	C35	2-FCV-90-115-B	CVI
	C41	2-FCV-43-54D-B	CIA
	C42	2-FCV-43-56D-B	CIA
	C43	2-FCV-43-59D-B	CIA
	C44	2-FCV-43-63D-B	CIA

CIA: CONTAINMENT ISOLATION PHASE A
CIB: CONTAINMENT ISOLATION PHASE B
CVI: CONTAINMENT VENT ISOLATION
SI: SAFETY INJECTION
*: No adverse impact on power supplies if energized after accident signal reset.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Inlet Door Position Monitoring System

BASES

BACKGROUND

Ninety-six limit switches monitor the position of the lower inlet doors. Two switches are mounted on the door frame for each door panel. The position and movement of the switches are such that the doors must be effectively sealed before the switches are actuated. A single annunciator window in the control room gives a common alarm signal when any door is open. Open/shut indication is also provided at the lower inlet door position display panel located in the Main Control Room. For door monitoring purposes, the ice condenser is divided into six zones, each containing four inlet door assemblies, or a total of eight door panels. The limit switches on the doors in any single zone are wired to a single light on the inlet door position display panel such that a closed light indicates that all the doors in that zone are shut and an open light indicates that one or more doors in that zone are open (Ref. 1). Additional information on the design of the lower inlet door monitoring instrumentation is provided in UFSAR Section 6.7.15.

Monitoring of inlet door position is necessary because the inlet doors form the barrier to air flow through the inlet ports of the ice condenser for normal unit operation. Failure of the Inlet Door Position Monitoring System requires an alternate OPERABLE monitoring system to be used to ensure that the ice condenser is not degraded.

APPLICABLE SAFETY ANALYSES

Proper operation of the inlet doors is necessary to mitigate the consequences of a loss of coolant accident or a main steam line break inside containment. The Inlet Door Position Monitoring System, however, is not required for proper operation of the inlet doors, nor is it considered OPERABLE as an initial condition for a DBA. Hence, the Inlet Door Position Monitoring System is not a consideration in the analyses of DBAs. Based on the PRA Summary Report in References 2 and 3, the Inlet Door Position Monitoring System has not been identified as a significant risk contributor.

(continued)