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

November 14, 2011

10 CFR 50.4

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

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

Subject: **CHANGES MADE TO THE TECHNICAL SPECIFICATION BASES**

The purpose of this letter is to provide the NRC with copies of changes that have been made to the Watts Bar Nuclear Plant (WBN), Unit 1 Technical Specification (TS) Bases through Revision 113. This information is provided in accordance with WBN TS Section 5.6, "TS Bases Control Program," on a frequency consistent with 10 CFR 50.71(e). These changes have been implemented at WBN during operating Cycle 10 and meet the criteria described within the TS Bases Control Program for which prior NRC approval is not required. The updates to the TS Bases are provided in the enclosures listed below.

There are no regulatory commitments in this submittal. Please direct any questions concerning this matter to Kara Stacy, Program Manager at (423) 751-3489.

Respectfully,

J. W. Shea

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

Cc: See Page 2

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NRR

Enclosures
cc (Enclosures):

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

Enclosure 1

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LIST OF ACRONYMS

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

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

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

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

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Revision 33	05-02-00	Clarification identifying four distribution boards primarily used for operational convenience.
Revision 34 (Amendment 24)	07-07-00	Elimination of Response Time Testing
Revision 35	08-14-00	Clarification of ABGTS Surveillance Testing
Revision 36 (Amendments 22 and 25)	08-23-00	Revision of Ice Condenser sampling and flow channel surveillance requirements
Revision 37 (Amendment 26)	09-08-00	Administrative Controls for Open Penetrations During Refueling Operations
Revision 38	09-17-00	SR 3.2.1.2 was revised to reflect the area of the core that will be flux mapped.
Revision 39 (Amendments 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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Revision 61 (Amendments 40 and 48)	10-14-03	Incorporated changes required to implement the Tritium Program (Amendment 40) and Stepped Boron Concentration increases for RWST and CLAs (Amendment 48) depending on the number of TPBARS installed into the reactor core.
Revision 62	10-15-03	Clarified ECCS venting in Bases Section B 3.5.2 (WBN-TS-03-19)
Revision 63	12-08-03	The contingency actions listed in Bases Table 3.8.1-2 were reworded to be consistent with the NRC Safety Evaluation that approved Tech Spec Amendment 39.
Revision 64 (Amendment 50)	03-23-04	Incorporated Amendment 50 for the seismic qualification of the Main Control Room duct work. Amendment 50 revised the Bases for LCO 3.7.10, "CREVS," and LCO 3.7.11, "CREATCS." An editorial correction was made on Page B 3.7-61.
Revision 65	04-01-04	Revised the Bases for Action B.3.1 of LCO 3.8.1 to clarify that a common cause assessment is not required when a diesel generator is made inoperable due to the performance of a surveillance.
Revision 66	05-21-04	Revised Page B 3.8-64 (Bases for LCO 3.8.4) to add a reference to SR 3.8.4.13 that was inadvertently deleted by the changes made for Amendment 12.
Revision 67 (Amendment 45)	03-05-05	Revised the Bases for LCOs 3.8.7, 3.8.8 and 3.8.9 to incorporate changes to the Vital Inverters (DCN 51370). Refer to the changes made for Bases Revision 58 (Amendment 45)
Revision 68 (Amendment 55)	03-22-05	Amendment 55 modified the requirements for mode change limitations in LCO 3.0.4 and SR 3.0.4 by incorporating TSTF-359, Revision 9.

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

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

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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.

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

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 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.

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 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.

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Enclosure 2

WBN, Unit 1 Technical Specification Bases - Changed Pages

BASES

SAFETY LIMIT VIOLATIONS (continued)

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).

2.2.4

If SL 2.1.1 is violated, the Plant Manager and Site Vice President shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.5

If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Reference 6). A copy of the report shall also be provided to the Plant Manager, Site Vice President, and Nuclear Safety Review Board.

2.2.6

If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 10, "Reactor Design."
2. Watts Bar FSAR, Section 7.2, "Reactor Trip System."
3. WCAP-8746-A, "Design Bases for the Overtemperature ΔT and the Overpower ΔT Trips," March 1977.

(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.

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.3

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 6).

2.2.4

If the RCS pressure SL is violated, the Plant Manager and Site Vice President shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 7). A copy of the report shall also be provided to the Plant Manager, Site Vice President, and Nuclear Safety Review Board.

2.2.6

If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary"; General Design Criterion 15, "Reactor Coolant System Design"; and General Design Criterion 28, "Reactivity Limits."

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure."

(continued)

BASES

REFERENCES (continued)

3. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, IWX-5000, "System Pressure Tests."

4. Title 10, Code of Federal Regulations, Part 100, "Reactor Site Criteria."

5. Watts Bar FSAR, Section 7.2, "Reactor Trip System."

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

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

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

BASES (continued)

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to

(continued)

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.

In cases where a non-TS support system has two redundant 100 percent capacity subsystems, each capable of supporting both TS trains, loss of one support system does not result in a loss of support for either train of TS equipment. Both TS systems remain operable, despite a loss of support function redundancy, because the TS definition of operability does not require a TS subsystem's necessary support function to meet the single failure criterion. No TS limits duration of the non-TS support subsystem outage, even though the single-failure design requirement of the supported TS system is not met. Appropriate duration for a maintenance activity should be determined by assessing and managing risk in accordance with paragraph (a)(4) of 10CFR50.65, the 'maintenance rule." If inoperable for more than 90 days, the licensee would have to evaluate the maintenance configuration as a change to the facility under 10CFR50.59, including consideration of the single failure design criterion.

(continued)

BASES (continued)

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the plant. These special tests and operations are necessary to demonstrate select plant performance characteristics, to perform special maintenance activities, and to perform special evolutions.

Test Exception LCOs 3.1.9 and 3.1.10 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

(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 the NRC Policy Statement.

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 OPERABILITY requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

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)-$T_{avg} \leq$ 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 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)

BASES

ACTIONS (continued)

A.2

If the untrippable 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

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

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." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

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)

BASES

ACTIONS

B.2.1.1 and B.2.1.2 (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.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

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 $F_Q(Z)$ 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. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

(continued)

ACTIONS (continued)

A.1

When one ARPI channel per group fails, the position of the rod can still be determined 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). Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.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.

A.2.1, A.2.2

The control rod drive mechanism (a portion of the rod control system) consists of four separate subassemblies; 1) the pressure vessel, 2) the coil stack assembly, 3) the latch assembly, and 4) the drive rod assembly. The coil stack assembly contains three operating coils; 1) the stationary gripper coil, 2) the moveable gripper coil, and 3) the lift coil. In support of Actions A.2.1 and A.2.2, a Temporary Alteration (TA) to the configuration of the plant is implemented to provide instrumentation for the monitoring of the rod control system parameters in the Main Control Room. The TA creates a circuit that monitors the operation and timing of the lift coil and the stationary gripper coil. Additional details regarding the TA are provided in the FSAR (Ref. 14).

Required Actions A.2.1 and A.1 are essentially the same. Therefore, the discussion provided above for Required Action A.1 applies to Required Action A.2.1. The options provided by Required Actions A.2.1 and A.2.2 allow for continued operation in a situation where the component causing the ARPI to be inoperable is inaccessible due to operating conditions (adverse radiological or temperature environment). In this situation, repair of the ARPI cannot occur until the unit is in an operating MODE that allows access to the failed components.

In addition to the initial 8 hour verification, Required Action A.2.1 also requires the following for the rod with the failed ARPI:

1. Verification of the position of the rod every 31 days using either the incore movable detectors or the PDMS.

2. Verification of the position of the rod using either the incore movable detectors or the PDMS within 8 hours of the performance of Required Action A.2.2 whenever there is an indication of unintended rod movement based on the parameters of the rod control system.

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BASES (continued)

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1 and B.2

When THERMAL POWER is > 85% RTP, the only acceptable actions are to reduce THERMAL POWER to ≤ 85% RTP or to suspend the PHYSICS TESTS exceptions. With the PHYSICS TESTS exceptions suspended, the PHYSICS TESTS may proceed if all other LCO requirements are met. Fuel integrity may be challenged with control rods or shutdown rods misaligned and THERMAL POWER > 85% RTP. The allowed Completion Time of 1 hour is reasonable, based on operating experience, for completing the Required Actions in an orderly manner and without challenging plant systems. This Completion Time is also consistent with the Required Actions of the LCOs that are suspended by the PHYSICS TESTS.

C.1 and C.2

When the Power Range Neutron Flux - High trip setpoints are > 10% RTP above the PHYSICS TESTS power level or > 90% RTP, the Reactor Trip System (RTS) may not provide the required degree of core protection if the trip setpoint is greater than the specified value.

The only acceptable actions are to restore the trip setpoint to the allowed value or to suspend the performance of the PHYSICS TESTS exceptions. The Completion Time of 1 hour is based on the practical amount of time it may take to restore the Neutron Flux - High trip setpoints to the correct value, consistent with operating plant safety. This Completion Time is consistent with the Required Actions of the LCOs that are suspended by the PHYSICS TESTS.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that the THERMAL POWER level is ≤ 85% RTP will ensure that the required core protection is provided during the performance of PHYSICS TESTS. Control of the reactor power level is a vital parameter and is closely monitored during the performance of PHYSICS TESTS. A Frequency of 1 hour is sufficient for ensuring that the power level does not exceed the limit.

SR 3.1.9.2

Verification of the Power Range Neutron Flux - High trip setpoints within 8 hours prior to initiation of the PHYSICS TESTS will ensure that the RTS is properly set to perform PHYSICS TESTS.

SR 3.1.9.3

The performance of SR 3.2.1.1 and SR 3.2.2.1 measures the core $F_Q(Z)$ and the $F^N_{\Delta H}$, respectively. If the requirements of these LCOs are met, the core has adequate protection from exceeding its design limits, while other LCO requirements are suspended. The Frequency of 12 hours is based on operating experience and the practical amount of time that it may take to obtain an incore power distribution measurement and calculate the hot channel factors.

SR 3.1.9.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.9.4 (continued)

f. Samarium concentration; and

g. Deleted

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

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

REFERENCES

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

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

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

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

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

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

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

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension and at high power. The PHYSICS TESTS for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed.

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions adjusted for uncertainty. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.7, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_Q(Z)$ is measured periodically using either the Movable Incore Detector System or the Power Distribution Monitoring System (PDMS) (Ref.6). These measurements are generally taken with the core at or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(Z)$ that are present during nonequilibrium situations, such as load following.

To account for these possible variations, the steady state value of $F_Q(Z)$ is adjusted by an elevation dependent factor that accounts for the calculated worst case transient conditions.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the peak cladding temperature 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. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

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

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

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

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

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

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

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

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

An $F_Q^C(Z)$ evaluation requires obtaining an incore power distribution measurement in MODE 1.

The measured value, $F^M{}_Q(Z)$, of $F_Q(Z)$ is obtained from the incore power distribution measurement and then corrected for fuel manufacturing tolerances and measurement uncertainty.

If the Moveable Incore Detector System is used to obtain the incore power distribution measurement, then:

$$F^C{}_Q(Z) = 1.03\ F^M{}_Q(Z)\ F^{MU}{}_Q$$

where 1.03 is the factor that accounts for the fuel manufacturing tolerances and $F^{MU}{}_Q$, which accounts for flux map measurement uncertainty, is 1.05 (Ref. 4).

When the PDMS is used to obtain the incore power distribution measurement, then:

$$F^C{}_Q(Z) = 1.03\ F^M{}_Q(Z)\ (1+U_Q/100)$$

(continued)

BASES

LCO (continued)

where 1.03 is the factor that accounts for the fuel manufacturing tolerances and the factor $(1+U_Q/100)$, which accounts for PDMS measurement uncertainty, is calculated and applied automatically by the BEACON™ software (Ref.6).

$F^C_Q(Z)$ is an approximation of the steady state $F_Q(Z)$.

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ W(Z)$$

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

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

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

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

APPLICABILITY

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

ACTIONS

A.1

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

(continued)

ACTIONS

A.2.1, A.2.2 (continued)

Required Action A.2.2 is in lieu of the verification of the position of the rod using either the incore movable detectors or PDMS every 8 hours as required by Required Action A.1. This action alleviates the potential for excessive wear on the incore system due to the repeated use of the incore detectors. Once the position of the rod with the failed ARPI is confirmed through the use of either the moveable incore detectors or PDMS in accordance with Required Action A.2.1, the parameters of the rod control system must be monitored until the failed ARPI is repaired. Should the review of the rod control system parameters indicate unintended movement of the rod, the position of the rod must be verified within 8 hours in accordance with Required Action A.2.1. Should there be unintended movement of the rod with the failed ARPI, an alarm will be received. Alarms will also be received if the rod steps in a direction other than what was demanded, and if the circuitry of the TA fails. Receipt of any alarm requires the verification of the position of the rod in accordance with Required Action A.2.1.

Required Actions A.2.1, A.2.2 and A.2.3 are modified by a note. The note clarifies that rod position monitoring by Required Actions A.2.1 and A.2.2 shall only be applied to one rod with an inoperable ARPI and shall only be allowed until the end of the current cycle. Further, Required Actions A.2.1, A.2.2 and A.2.3 shall not be allowed after the plant has been in MODE 5 or other plant condition, for a sufficient period of time, in which the repair of the inoperable ARPI(s) could have safely been performed.

As indicated previously, the modifications required for the monitoring of the rod control system will be implemented as a TA. Implementation of the TA includes a review for the impact on plant procedures and training. This ensures that changes are initiated for key issues like the monitoring requirements in the control room, and operator training on the temporary equipment.

A.2.3

Required Action A.2.3 addresses two contingency measures when the TA is utilized:

1. Verification of the position of the rod with the inoperable ARPI by use of either the moveable incore detectors or PDMS, whenever the rod is moved greater than 12 steps in one direction.

2. Operation of the unit when THERMAL POWER is less than or equal to 50% RTP.

For the first contingency, the rod group alignment limits of LCO 3.1.5 require that all shutdown and control rods be within 12 steps of their group step counter demand position. The limits on shutdown or control rod

(continued)

ACTIONS

A.2.3 (continued)

alignments ensure that the assumptions in the safety analysis will remain valid and that the assumed reactivity will be available to be inserted for a unit shutdown. Therefore, this conservative measure ensures LCO 3.1.5 is met whenever the rod with the inoperable ARPI is moved greater than 12 steps. For the second contingency, the reduction of THERMAL POWER to less than or equal to 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 13). Consistent with LCO 3.0.4 and this action, unit startup and operation to less than or equal to 50% RTP may occur with one ARPI per group inoperable. However, prior to escalating THERMAL POWER above 50% RTP, the position of the rod with an inoperable ARPI must be verified by use of either the moveable incore detectors or PDMS. Once 100% RTP is achieved, the position of the rod must be reverified within 8 hours by use of either the moveable incore detectors or PDMS. Monitoring of the rod control system parameters in accordance with Required Action A.2.2 for the rod with an inoperable ARPI may resume upon completion of the verification at 100% RTP.

A.3

Required Action A.3 applies whenever the TA is not utilized. The discussion for Required Action A.2.3 (above) clarified that a reduction of THERMAL POWER to less than or equal to 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 13). The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to less than or equal to 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above. Consistent with LCO 3.0.4 and this action, unit startup and operation to less than or equal to 50% RTP may occur with one ARPI per group inoperable. Thermal Power may be escalated to 100% RTP as long as Required Action A.1 is satisfied.

B.1 and B.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to less than or equal to 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at greater than 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.

(continued)

BASES

ACTIONS (continued)

C.1.1 and C.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the ARPI 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 less than or equal to 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

C.2

Reduction of THERMAL POWER to less than or equal 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 C.1.1 and C.1.2 or reduce power to less than or equal to 50% RTP.

D.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 ARPI agrees with the demand position within 12 steps ensures that the ARPI is operating correctly.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for unnecessary plant transients if the SR were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at a Frequency of once every 18 months. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."
2. Watts Bar FSAR, Section 15.2.1, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition."
3. Watts Bar FSAR, Section 15.2.2, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power."
4. Watts Bar FSAR, Section 15.2.3, "Rod Cluster Control Assembly Misalignment."
5. Watts Bar FSAR, Section 15.2.4, "Uncontrolled Boron Dilution."
6. Watts Bar FSAR, Section 15.2.5, "Partial Loss of Forced Reactor Coolant Flow."
7. Watts Bar FSAR, Section 15.2.13, "Accidental Depressurization of the Main Steam System."
8. Watts Bar FSAR, Section 15.3.4, "Complete Loss of Forced Reactor Coolant Flow."
9. Watts Bar FSAR, Section 15.3.6, "Single Rod Cluster Control Assembly Withdrawal At Full Power."
10. Watts Bar FSAR, Section 15.4.2.1, "Major Rupture of Main Steam Line."
11. Watts Bar FSAR, Section 15.4.4, "Single Reactor Coolant Pump Locked Rotor."
12. Watts Bar FSAR, Section 15.4.6, "Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)."
13. Watts Bar FSAR, Section 4.3, "Nuclear Design."
14. Watts Bar FSAR, Section 7.7.1.3.2, "Main Control Room Rod Position Indication."
15. WCAP-12472-P-A,"BEACON Core Monitoring and Operations Support System," August 1994.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

SR 3.2.1.1

Verification that $F_Q^C(Z)$ is within its specified limits involves increasing $F_Q^M(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_Q^C(Z)$. Specifically, $F_Q^M(Z)$ is the measured value of $F_Q(Z)$ obtained from an incore power distribution measurement.

If the Movable Incore Detector System is used to obtain the incore power distribution measurement, then:

$$F^C{}_Q(Z) = 1.03\ F^M{}_Q(Z)\ F^{MU}{}_Q$$

where 1.03 is the factor that accounts for the fuel manufacturing tolerances and $F^{MU}{}_Q$, which accounts for flux map measurement uncertainty, is 1.05 (Ref. 4).

When the PDMS is used to obtain the incore power distribution measurement, then:

$$F^C{}_Q(Z) = 1.03\ F^M{}_Q(Z)\ (1+U_Q/100)$$

where 1.03 is the factor that accounts for the fuel manufacturing tolerances and the factor $(1+U_Q/100)$, which accounts for PDMS measurement uncertainty, is calculated and applied automatically by the BEACON software (Ref.6)

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

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

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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

SR 3.2.1.2

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

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

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

a. Lower core region, from 0 to 10% inclusive; and

b. Upper core region, from 90 to 100% inclusive.

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

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

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

maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)}\right]$,

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

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

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

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

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

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
2. Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized water Reactors," May1974.

(continued)

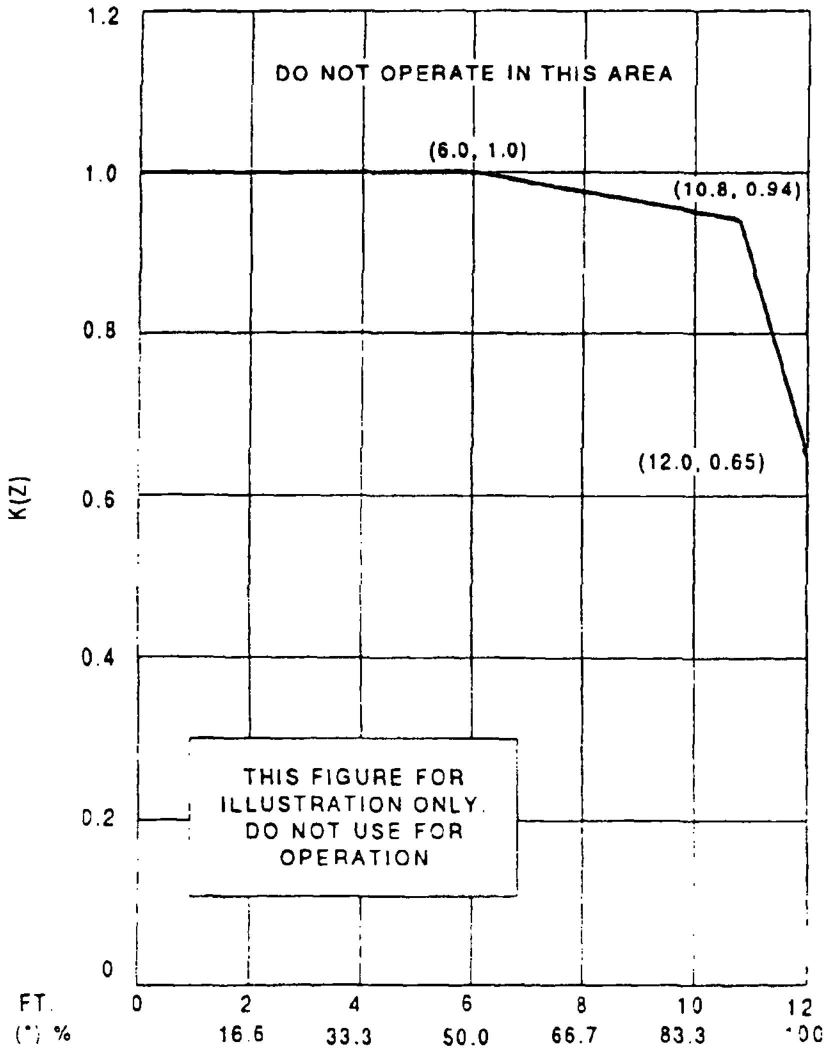
REFERENCES (continued)

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

4. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.

5. Westinghouse Technical Bulletin (TB) 08-4, "F_Q Surveillance at Part Powers," July 15, 2008.

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



CORE HEIGHT

*For core height of 12 feet

Figure B 3.2.1-1 (page 1 of 1)
K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor $(F^N_{\Delta H})$

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from an incore power distribution measurement obtained with the Movable Incore Detector System or the Power Distribution Monitoring System (PDMS). Specifically, the results of the three dimensional incore power distribution measurements are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB for the hottest fuel rod in the core. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements.

(continued)

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 is applied in the heated region below the first mixing vane grid. In addition, the W-3 DNB correlation is 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Application of these criteria provides assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use this variable value of $F^N_{\Delta H}$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an initial $F^N_{\Delta H}$ as a function of power level defined by the COLR limit equation.

The LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature (Ref. 3) model $F^N_{\Delta H}$ as well as the Nuclear Heat Flux Hot Channel Factor ($F_Q(Z)$).

The fuel is protected in part by Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.7, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

$F^N_{\Delta H}$ and $F_Q(Z)$ are measured periodically using either the Movable Incore Detector System or the PDMS (Ref. 5). Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F^N_{\Delta H}$ satisfies Criterion 2 of the NRC Policy Statement.

LCO

$F^N_{\Delta H}$ shall be maintained within the limits of the relationship provided in the COLR.

The $F^N_{\Delta H}$ limit identifies the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB.

(continued)

BASES

ACTIONS (continued)

A.2

Once the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore power distribution measurement (SR 3.2.2.1) must be obtained and the measured value of $F^N_{\Delta H}$ verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore power distribution measurement, perform the required calculations, and evaluate $F^N_{\Delta H}$.

A.3

Verification that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The value of $F^N_{\Delta H}$ is determined by using either the Movable Incore Detector System or the PDMS to obtain an incore power distribution measurement. A data reduction computer program then calculates the maximum value of $F^N_{\Delta H}$ from the measured flux distributions. The measured value of $F^N_{\Delta H}$ must be multiplied by a factor to account for measurement uncertainty before making comparisons to the $F^N_{\Delta H}$ limit.

If the Moveable Incore Detector System is used to obtain the incore power distribution measurement, then the factor ($F^{MU}_{\Delta H}$) is 1.04 (Ref. 4). When the PDMS is used to obtain the incore power distribution measurement, the factor ($1+U_{\Delta H}/100$) is calculated and applied automatically by the BEACON software (Ref. 5).

After the initial fuel loading and each refueling, $F^N_{\Delta H}$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F^N_{\Delta H}$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

2. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

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

4. WCAP-7308-L-P-A, “Evaluation of Nuclear Hot Channel Factor Uncertainties,” June 1988.

5. WCAP-12472-P-A, “BEACON Core Monitoring and Operations Support System,” August 1994.

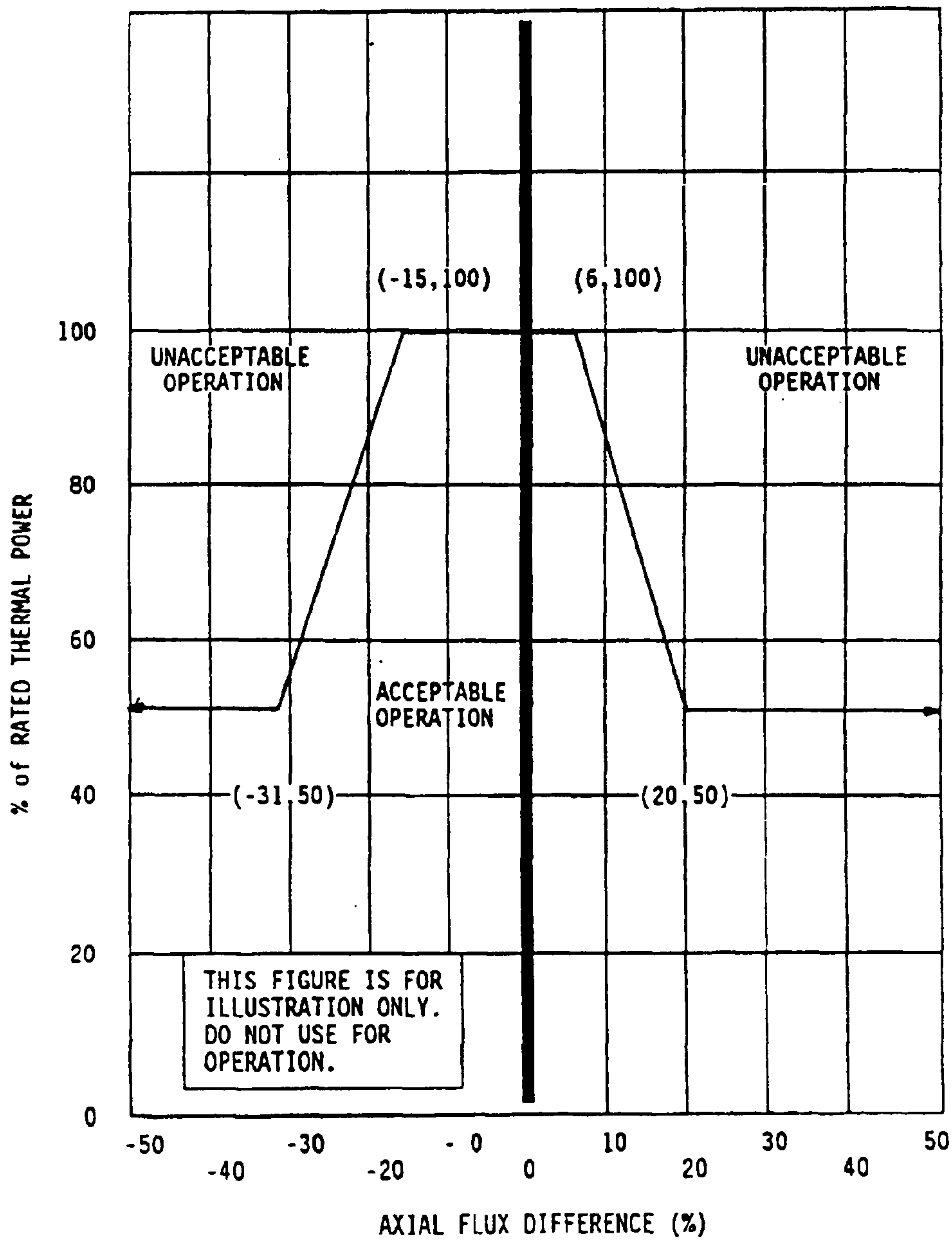


Figure B 3.2.3-1
TYPICAL AXIAL FLUX DIFFERENCE Acceptable Operation Limits as a Function of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), rod group alignment, sequence, overlap, and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and ($F^N_{\Delta H}$) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

(continued)

ACTIONS (continued)

A.2

After completion of Required Action A.1, the QPTR Alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform an incore power distribution measurement. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta}H$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.4.2 (continued)

For the purpose of monitoring the QTPR 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)

BASES

REFERENCES

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

2. Regulatory Guide 1.77, Rev. 0, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

3. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability."

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

ACTIONS (continued)

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

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

The NIS power range detectors provide input to the CRD System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 14.

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to ≤ 75% RTP within 78 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one of the NIS power range detectors inoperable, 1/4 of the radial power distribution monitoring capability is lost.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within 72 hours and the QPTR monitored once every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels ≥ 75% RTP. The 12 hour Frequency is consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

As an alternative to the above actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Seventy-eight hours are allowed to place the plant in MODE 3. The 78-hour Completion Time includes 72 hours for channel corrective maintenance and an additional 6 hours for the MODE reduction as required by Required Action D.3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 12 hours while performing routine surveillance testing of other channels. The Note also allows placing

(continued)

ACTIONS

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

the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 12 hour time limit is justified in Reference 14.

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

E.1 and E.2

Condition E applies to the following reactor trip Functions:

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

A known inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 14.

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

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

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

SR 3.3.1.2

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

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

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

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3

SR 3.3.1.3 compares the incore power distribution measurement to the NIS channel AFD output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function. The incore power distribution measurement may be obtained using the Movable Incore Detector System or an OPERABLE Power Distribution Monitoring System (PDMS) (Ref. 16).

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

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

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 62 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.4 (continued)

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

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 92 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection Function. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 15.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore power distribution measurement(s). If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the Overtemperature ΔT Function. The incore power distribution measurement(s) may be obtained using the Movable Incore Detector System or an OPERABLE PDMS (Ref. 16).

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

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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7

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

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

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

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

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

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

SR 3.3.1.8

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 *(continued)*

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" (Reference 12), provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

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

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

REFERENCES

1. Watts Bar FSAR, Section 6.0, "Engineered Safety Features"
2. Watts Bar FSAR, Section 7.0, "Instrumentation and Controls"
3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.
5. 10 CFR Part 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."

(continued)

REFERENCES (continued)

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

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

8. Watts Bar Technical Requirements Manual, Section 3.3.1, "Reactor Trip System Response Times."

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

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

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

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

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

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

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

16. WCAP-12472-P-A,“BEACON Core Monitoring and Operations Support System,” August 1994.

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B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

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

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

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

<u>Field Transmitters or Sensors</u>

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

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. Auxiliary Feedwater-Trip Of All Main Feedwater Pumps

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

This Function must be OPERABLE in MODES 1 and 2 in accordance with the applicable Tech Specs Notes to ensure that at least one SG is provided with sufficient water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident.

During plant startup in Mode 2 at approximately 4% rated thermal power, a MFPT is placed in service along with the operating AFW pump(s). During the process of placing the first MFPT in service, the anticipatory AFW auto-start channel for the non-operating MFPT is placed in "bypass" (i.e., the MFPT control circuit is de-energized). This action is necessary to prevent inadvertent AFW auto-start during pump turbine rollup and/or overspeed trip testing. Inadvertent AFW actuation (i.e., startup of all three AFW pumps) under no load conditions could result in an overcooling transient due to the steam flow demand by the turbine driven AFW pump and failure of the AFW LCVs to throttle back quickly enough to maintain programmed SG level. The consequences would be unnecessary SG blowdown isolation, a potential positive reactivity transient, and unnecessary NRC reports for ESF actuations.

With two motor driven AFW pumps (or one turbine driven AFW pump) running in parallel with one MFPT, SG level control is transferred from the AFW level control valves (LCVs) to the MFW Bypass Regulating (Reg) Valves. During this transition it is not necessary nor is it required that the anticipatory AFW auto-start circuit be OPERABLE. This is because the AFW system under these conditions is already in service and will automatically realign if an ESF actuation were to occur (Ref. 20).

Upon completion of the transfer of SG level control to the MFW Bypass Reg Valves, the motor driven AFW pumps or the turbine driven AFW pump will be on recirc flow, and the operating MFPT

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

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

will be providing feed flow to all four SG loops. When these conditions are met, the operator will reset the AFW auto-start circuit by performing the following actions:

1. Transfer the anticipatory AFW auto-start channel for the non-operating MFPT from the "bypass" position to the "trip" position (i.e., MFPT control circuit is energized and the pump turbine hydraulic control circuit de-pressurized), and then;

2. Secure the AFW pumps in standby mode.

Under these conditions, the anticipatory AFW auto-start circuit will be in a half trip condition (one-out-of-two) in Mode 2 and during transitions from Mode 2 to Mode 1. If the operating MFPT were to trip during this time period, an AFW auto-start signal would be generated causing all three AFW pumps to start.

During plant shutdown in Mode 2, the sequence of events described above is reversed i.e., the AFW pump(s) are placed in service along with the operating MFPT. After SG level control has been transferred from the MFW Bypass Reg Valves to the AFW LCVs the anticipatory AFW auto-start circuit is disabled and the MFPT secured.

Mode 1 applicability allows entry into LCO 3.3.2, Condition J to be suspended for up to 4 hours when placing the second MFPT in service or removing one of two MFPTs from service. This provision will reduce administrative burden on the plant. Plant safety is not compromised during this short period because the safety grade AFW auto-start channels associated with steam generator low-low levels are operable.

In Mode 3, decay heat and sensible heat removal is sufficiently low that adequate time is available for the operator to manually activate the AFW system if a loss of MFW were to occur.

In Modes 4, and 5; the RCPs and MFPTs are normally shut down with decay heat and sensible heat removed by the RHR system. Therefore, neither pump trip is indicative of a condition requiring automatic AFW initiation.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

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

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

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

7. Automatic Switchover to Containment Sump

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

(continued)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.2 (continued)

each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference 18.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 92 days on a STAGGERED TEST BASIS. The Frequency of 92 days is justified in Reference 18.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.2-1.

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

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.2.4 (continued)

The Frequency of 184 days is justified in Reference 18, except for Function 7. The Frequency for Function 7 is justified in References 10 and 18.

SR 3.3.2.5

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

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

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

SR 3.3.2.6

SR 3.3.2.6 is the performance of a TADOT every 92 days. This test is a check of the AFW Pumps Train A and B Suction Transfer on Suction Pressure—Low (Function 6.f), and Turbine Trip and Feedwater Isolation - Main Steam Valve Vault Rooms Water Level - High (Function 5.d and 5.e).

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.3

SR 3.3.5.3 is the performance of a CHANNEL CALIBRATION.

The setpoints, as well as the response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay, as shown in Reference 1.

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

The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. Watts Bar FSAR, Section 8.3, "Onsite (Standby) Power System."
2. Watts Bar FSAR, Section 15.0, "Accident Analysis."
3. TVA Calculation WPE2119202001, "6.9 kV Shutdown and Logic Boards Undervoltage Relays Requirements/Demonstrated Accuracy Calculation."
4. *Technical Requirements Manual, Section* 3.3.2, "Engineered *Safety* Feature Response Times."
5. TVA Calculation TDR SYS.211-LV1, "Demonstrated Accuracy Calculation TDR SYS.211-LV1."
6. TVA Calculation TDR SYS.211-DS1, "Demonstrated Accuracy Calculation TDR SYS.211-DS1."

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Vent Isolation Instrumentation

BASES

BACKGROUND

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

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

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

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

The plant design basis requires that when moving irradiated fuel in the Auxiliary Building and/or Containment with the Containment open to the Auxiliary Building ABSCE spaces, a signal from the spent fuel pool radiation monitors 0-RE-90-102 and 103 will initiate a Containment Ventilation Isolation (CVI) in addition to their normal function. In addition, a signal from the containment purge radiation monitors 1 -RE-90-130, and -13 1 or other CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required.

As stated previously, provisions are provided to interconnect the ABI* and the CVI instrumentation to facilitate the actuation of an ABI* when a CVI is initiated and the actuation of a CVI when an ABI* is initiated. The following table specifies the radiation monitors and the associated actuation instrumentation that must be available when the ABI* and CVI functions are interconnected and fuel is being moved either inside containment or in the Auxiliary Building or if fuel movement is taking place in both areas:

Fuel Movement:	Required Radiation Monitors:	Required Actuation Instrumentation:	Containment Hatch:	Containment Penetrations:
Inside Containment	1-RE-90-130 1-RE-90-131	LCO 3.3.6 LCO 3.3.8	Closed in accordance with LCO 3.9.4	Containment open to Auxiliary Building in accordance with LCO 3.9.4
Auxiliary Building	0-RE-90-102 0-RE-90-103	LCO 3.3.6 LCO 3.3.8	Open or Closed	ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12
Inside Containment and in Auxiliary Building	1-RE-90-130 1-RE-90-131 0-RE-90-102 0-RE-90-103	LCO 3.3.6 LCO 3.3.8	Closed in accordance with LCO 3.9.4	In accordance with LCO 3.9.4 ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12

* That portion of the ABI initiated by the Spent Fuel Pool Radiation Monitors, 0-RE-90-102 and 0-RE-90-103.

(continued)

BASES (continued)

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Vent Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment Vent Isolation at any time by using either of two switches in the control room or from local panel(s). Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals. These manual switches also initiate a Phase A isolation signal.

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

Each channel consists of one selector switch and the interconnecting wiring to the actuation logic cabinet.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable MODES and specified conditions for the containment vent isolation portion of the SI Function is different and less restrictive than those for the SI role. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Vent Isolation Function is affected, the Conditions applicable to the SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Vent Isolation Functions specify sufficient compensatory measures for this case.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8 Auxiliary Building Gas Treatment (ABGTS) Actuation Instrumentation

BASES

BACKGROUND

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

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

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required.

As stated previously, provisions are provided to interconnect the ABI* and the CVI instrumentation to facilitate the actuation of an ABI* when a CVI is initiated and the actuation of a CVI when an ABI* is initiated. The following table specifies the radiation monitors and the associated actuation instrumentation that must be available when the ABI* and CVI functions are interconnected and fuel is being moved either inside containment or in the Auxiliary Building or if fuel movement is taking place in both areas:

Fuel Movement:	Required Radiation Monitors:	Required Actuation Instrumentation:	Containment Hatch:	Containment Penetrations:
Inside Containment	1-RE-90-130 1-RE-90-131	LCO 3.3.6 LCO 3.3.8	Closed in accordance with LCO 3.9.4	Containment open to Auxiliary Building in accordance with LCO 3.9.4
Auxiliary Building	0-RE-90-102 0-RE-90-103	LCO 3.3.6 LCO 3.3.8	Open or Closed	ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12
Inside Containment and in Auxiliary Building	1-RE-90-130 1-RE-90-131 0-RE-90-102 0-RE-90-103	LCO 3.3.6 LCO 3.3.8	Closed in accordance with LCO 3.9.4	In accordance with LCO 3.9.4 ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12

* That portion of the ABI initiated by the Spent Fuel Pool Radiation Monitors, 0-RE-90-102 and 0-RE-90-103.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The maximum dose to the whole body and the thyroid that an individual occupying the Main Control Room can receive for the accident duration is specified in 10 CFR 50, Appendix A, GDC 19. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits and within the 10 CFR 50, Appendix A, GDC 19 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite and Main Control Room radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) or main steam line break (MSLB) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits, and ensure the Main Control Room accident dose is within the appropriate 10 CFR 50, Appendix A, GDC 19 dose guideline limits.

The evaluations showed the potential offsite and Main Control Room dose levels for a SGTR and MSLB accident were within the appropriate 10 CFR 100 and GDC 19 guideline limits.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary and Main Control Room accident doses will not exceed the appropriate 10 CFR 100 dose guideline limits and 10 CFR 50, Appendix A, GDC 19 dose guideline limits following a SGTR or MSLB accident. The SGTR and MSLB safety analysis (Ref. 2) assume the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gallons per day (GPD). The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.14, "Secondary Specific Activity."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR and MSLB accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analyses are for two cases of reactor coolant specific activity. One case assumes specific activity at 0.265 μCi/gm DOSE EQUIVALENT I-131 with an iodine spike immediately after the accident that increases the iodine activity in the reactor coolant by a factor of 500 times the iodine production rate necessary to maintain a steady state iodine concentration of 0.265 μCi/gm DOSE EQUIVALENT I-131. The second case assumes the initial reactor coolant iodine activity at 21 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant equals the LCO limit of 100/ $\bar{E}$ μCi/gm for gross specific activity.

Regarding the specific activity values for DOSE EQUIVALENT I-131 defined above, the Functional Evaluation for Problem Evaluation Report 360041 (Ref. 3) documents that the pre-accident iodine spike of 21 μCi/gm is incorrect and that the correct limit is 14 μCi/gm. The 14 μCi/gm limit is being administratively controlled through Step 5 and 6 of Section 3.0, "Operator Actions," of AOI-28, "High Activity in Reactor Coolant," and the note preceding Step 5 until NRC's approval of a License Amendment Request updating Technical Specification 3.4.16 can be obtained.

The analysis also assumes a loss of offsite power at the same time as the SGTR and MSLB event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. The MSLB results in a reactor trip due to low steam pressure.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of a SGTR and MSLB accident are within the appropriate 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 21 μCi/gm DOSE EQUIVALENT I-131, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 21 μCi/gm DOSE EQUIVALENT I-131.

Regarding the specific activity value for DOSE EQUIVALENT I-131 defined above, the Functional Evaluation for Problem Evaluation Report 360041 (Ref. 3) documents that the pre-accident iodine spike of 21 μCi/gm is incorrect and that the correct limit is 14 μCi/gm. The 14 μCi/gm limit is being administratively controlled through Step 5 and 6 of Section 3.0, "Operator Actions," of AOI-28, "High Activity in Reactor Coolant," and the note preceding Step 5 until NRC's approval of a License Amendment Request updating Technical Specification 3.4.16 can be obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specific iodine activity is limited to 0.265 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary and accident dose to personnel in the Main Control Room during the Design Basis Accident (DBA) will be within the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary and accident dose to personnel in the Main Control Room during the DBA will be within the allowed whole body dose.

The SGTR and MSLB accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels and Main Control Room accident dose are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SGTR or MSLB, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits, or Main Control Room accident dose that exceed the 10 CFR 50, Appendix A, GDC 19 dose limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an accident to within the acceptable Main Control Room and site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limit of 21 μCi/gm is not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

Regarding the specific activity value for DOSE EQUIVALENT I-131 defined above, the Functional Evaluation for Problem Evaluation Report 360041 (Ref. 3) documents that the pre-accident iodine spike of 21 μCi/gm is incorrect and that the correct limit is 14 μCi/gm. The 14 μCi/gm limit is being administratively controlled through Step 5 and 6 of Section 3.0, "Operator Actions," of AOI-28, "High Activity in Reactor Coolant," and the note preceding Step 5 until NRC's approval of a License Amendment Request updating Technical Specification 3.4.16 can be obtained.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1 and B.2

With the gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is greater than 21 μCi/gm, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

Regarding the specific activity value for DOSE EQUIVALENT I-131 defined above, the Functional Evaluation for Problem Evaluation Report 360041 (Ref. 3) documents that the pre-accident iodine spike of 21 μCi/gm is incorrect and that the correct limit is 14 μCi/gm. The 14 μCi/gm limit is being administratively controlled through Step 5 and 6 of Section 3.0, "Operator Actions," of AOI-28, "High Activity in Reactor Coolant," and the note preceding Step 5 until NRC's approval of a License Amendment Request updating Technical Specification 3.4.16 can be obtained.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following rapid power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," 1973.
2. Watts Bar FSAR, Section 15.4, "Condition IV - Limiting Faults."
3. Functional Evaluation for Problem Evaluation Report 360041 (T35 110504 809)

BASES

APPLICABLE SAFETY ANALYSES (continued)

heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY (Ref. 3).

The CREATCS satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two independent and redundant trains of the CREATCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CREATCS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the chillers, AHUs, and associated temperature control instrumentation. In addition, the CREATCS must be operable to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies, the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

In MODE 5 or 6, CREATCS is required during a control room isolation following a waste gas decay tank rupture.

ACTIONS

A.1

With one CREATCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREATCS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CREATCS train could result in loss of CREATCS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

(continued)

ACTIONS (continued)

A.1

During modification activities to replace the CREATCS chillers, an allowance is permitted for one CREATCS train to be inoperable for 60 days provided the following compensatory measures are in place:

A temporary chilled water package will be installed and maintained in a "standby" condition. During the initial installation, the chiller skids and chilled water pumps will be stationed in the yard with the chilled water lines filled and vented at the manifolds in the Control Building Mechanical Equipment Room located on Elevation 755.0 and the Shutdown Board Room Mechanical Equipment Rooms located on Elevation 757.0. Final connection of the chilled water hoses to the SDBR or MCR AHUs will not occur until that particular HVAC train is taken out of service for chiller replacement. All necessary hardware, hoses, and fittings will be stationed at the AHUs for rapid deployment in order to connect, fill and vent the temporary chilled water hoses to the AHUs. Procedures will be provided as part of Work Order documents to include instructions for startup, operation, preventative maintenance, and shutdown of the temporary cooling equipment. Qualified personnel will be provided training on these procedures. Furthermore, to provide additional defense-in-depth, the following requirements would also be implemented:

1. If a temporary chilled water system hose breaks in the Control Building during the timeframe that the temporary equipment is installed, the two manual isolation ball valves located at the MCRHZ boundary will be closed immediately. Qualified personnel will be capable of closing the valves and will be stationed in the area whenever the valves are in the "Open" position and the temporary cooling system is in service.

2. If a temporary chilled water system hose breaks in the Auxiliary Building or Shutdown Board Room during the timeframe that the temporary equipment is installed, the manual isolation valves will be closed immediately. Qualified personnel will be capable of closing the valves and will be stationed in the area whenever the valves are in the "Open" position and the temporary cooling system is in service.

3. Due to lack of operating data, the availability or reliability of the new MCR chiller packages are unknown. To compensate for this uncertainty the new train-B chiller packages will be operated for a minimum of 2 weeks prior to removing the train-A MCR chillers from service for replacement.

4. During replacement of the MCR chillers, no planned maintenance activity, except for SRs 3.8.1.2, 3.8.1.3, and 3.8.1.7 that could impact the OPERABILITY of the DG's that provide emergency power to the OPERABLE MCR chiller train will be performed.

This TS provision is only authorized for one entry per train during modification activities planned for the upgrade of the MCR chillers beginning no earlier than March 1, 2011 and ending April 30, 2012.

(continued)

ACTIONS (continued)

B.1 and B.2

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

C.1 and C.2

In MODE 5 or 6, or during movement of irradiated fuel, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

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

E.1

If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4 the CREATCS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

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SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the sizing calculations in the control room. This SR consists of a combination of testing and calculations. This is accomplished by verifying that the system has not degraded. The only measurable parameters that could degrade undetected during normal operation are the system air flow and chilled water flow rate. Verification of these two flow rates will provide assurance that the heat removal capacity of the system is still adequate. The 18 month Frequency is appropriate since significant degradation of the CREATCS is slow and is not expected over this time period.

REFERENCES

1. Watts Bar FSAR, Section 9.4.1, "Control Room Area Ventilation System."

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

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

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.12 Auxiliary Building Gas Treatment System (ABGTS)

BASES

BACKGROUND

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

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

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

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

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ABGTS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the ABGTS. The DBA analysis of the fuel handling accident assumes that only one train of the ABGTS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the ABSCE is determined for a fuel handling accident and for a LOCA. The assumptions and the analysis for a fuel handling accident follow the guidance provided in Regulatory Guide 1.25 (Ref. 5) and NUREG/CR-5009 (Ref. 11). The assumptions and analysis for a LOCA follow the guidance provided in Regulatory Guide 1.4 (Ref. 6).

The ABGTS satisfies Criterion 3 of the NRC Policy Statement.

The ABGTS is required to be operable during movement of irradiated fuel in the Auxiliary Building during any mode and during movement of irradiated fuel in the Reactor Building when the Reactor Building is established as part of the ABSCE boundary (see TS 3.3.8, 3.7.12, & 3.9.4). When moving irradiated fuel inside containment, at least one train of the containment purge system must be operating or the containment must be isolated. When moving irradiated fuel in the Auxiliary Building during times when the containment is open to the Auxiliary Building ABSCE spaces, containment purge can be operated, but operation of the system is not required.

As stated previously, provisions are provided to interconnect the ABI* and the CVI instrumentation to facilitate the actuation of an ABI* when a CVI is initiated and the actuation of a CVI when an ABI* is initiated. The following table specifies the radiation monitors and the associated actuation instrumentation that must be available when the ABI* and CVI functions are interconnected and fuel is being moved either inside containment or in the Auxiliary Building or if fuel movement is taking place in both areas:

Fuel Movement:	Required Radiation Monitors:	Required Actuation Instrumentation:	Containment Hatch:	Containment Penetrations:
Inside Containment	1-RE-90-130 1-RE-90-131	LCO 3.3.6 LCO 3.3.8	Closed in accordance with LCO 3.9.4	Containment open to Auxiliary Building in accordance with LCO 3.9.4
Auxiliary Building	0-RE-90-102 0-RE-90-103	LCO 3.3.6 LCO 3.3.8	Open or Closed	ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12
Inside Containment and in Auxiliary Building	1-RE-90-130 1-RE-90-131 0-RE-90-102 0-RE-90-103	LCO 3.3.6 LCO 3.3.8	Closed in accordance with LCO 3.9.4	In accordance with LCO 3.9.4 ABSCE maintained in accordance with LCO 3.3.8 and 3.7.12

* That portion of the ABI initiated by the Spent Fuel Pool Radiation Monitors, 0-RE-90-102 and 0-RE-90-103.

(continued)

BASES (continued)

LCO

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

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

a. Fan is OPERABLE;

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

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

APPLICABILITY

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

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

During movement of irradiated fuel in the fuel handling area, the ABGTS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. See additional discussion in the Background and Applicable Safety Analysis sections.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.3.3 (continued)

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity (≥ 190 psig, value does not account for instrument error, Ref. 7) for each DG is available. The system design requirements provide for a minimum of five engine start cycles without recharging. A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed. The pressure specified in this SR is intended to reflect the lowest value at which the five starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.3.5 (continued)

The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

SR 3.8.3.6

This SR performs visual inspection, in lieu of the 10 year hydrostatic test indicated in Regulatory Guide 1.137 Position C.1.e(1), of all exposed fuel oil piping while the diesel is running. Identified leakage does not constitute failure of this surveillance. Upon discovery, leakage is entered into the corrective action program and evaluated for impact on diesel generator operability and corrected as appropriate. The 18 month Frequency is based on engineering judgment and is consistent with the refueling cycle testing performed on the DGs.

SR 3.8.3.7

Draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are required at 10 year intervals by Regulatory Guide 1.137 (Ref. 2), paragraph 2.f. To preclude the introduction of surfactants in the fuel oil system, the cleaning should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents. This SR is for preventive maintenance. The presence of sediment does not necessarily represent a failure of this SR, provided that accumulated sediment is removed during performance of the Surveillance.

REFERENCES

1. Watts Bar FSAR, Section 8.3, "Onsite (Standby) Power System".
2. Regulatory Guide 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October, 1979.
3. ANSI N195-1976, "Fuel Oil Systems for Standby Diesel Generators," Appendix B.
4. Watts Bar FSAR, Section 9.5.7, "Diesel Engine Lubrication System."

(continued)

BASES

REFERENCES (continued)

5. Watts Bar FSAR, Section 15, "Accident Analysis" and Section 6 "Engineered Safety Features."

6. ASTM Standards:
D4057-1988, "Practice for Manual Sampling of Petroleum and Petroleum Products."
D975-1990, "Standard Specification for Diesel Fuel Oils."
D4176-1986, "Free Water and Particulate Contamination in Distillate Fuels."
D1552-1990, "Standard Test Method for Sulfur in Petroleum Products (High Temperature Method)."
D2622-1987, "Standard Test Method for Sulfur in Petroleum Products (X-Ray Spectrographic Method)."
D2276-1989, "Standard Test Method for Particulate Contamination in Aviation Fuel."
D1298-1985, "Standard Test Method for Density, Specific Gravity, or API Gravity of Crude Petroleum and Liquid Petroleum Products by Hydrometer Method."

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

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

<u>125 V Vital DC Electrical Power Subsystem</u>

The vital 125 VDC electrical power system is a Class IE system whose safety function is to provide control power for engineered safety features equipment, emergency lighting, vital inverters, and other safety-related DC powered equipment for the entire unit. The system capacity is sufficient to supply these loads and any connected nonsafety loads during normal operation and to permit safe shutdown and isolation of the reactor for the "loss of all AC power" condition. The system is designed to perform its safety function subject to a single failure.

The 125V DC vital power system is composed of the four redundant channels (Channels I and III are associated with Train A and Channels II and IV are associated with Train B) and consists of four lead-acid-calcium batteries, eight battery chargers (including two pairs of spare chargers), four distribution boards, battery racks, and the required cabling, instrumentation and protective features. Each channel is electrically and physically independent from the equipment of all other channels so that a single failure in one channel will not cause a failure in another channel. Each channel consists of a battery charger which supplies normal DC power, a battery for emergency DC power, and a battery board which facilitates load grouping and provides circuit protection. These four channels are used to provide emergency power to the 120V AC vital power system which furnishes control power to the reactor protection system. No automatic connections are used between the four redundant channels.

Battery boards I, II, III, and IV have a charger normally connected to them and also have manual access to a spare (backup) charger for use upon loss of the normal charger.

(continued)

BACKGROUND

125 V Vital DC Electrical Power Subsystem (continued)

Additionally, battery boards I, II, III, and IV have manual access to the fifth vital battery system. The fifth 125V DC Vital Battery System is intended to serve as a replacement for any one of the four 125V DC vital batteries during their testing, maintenance, and outages with no loss of system reliability under any mode of operation.

Each of the vital DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 6.9 kV switchgear, and 480 V load centers. The vital DC electrical power subsystems also provide DC electrical power to the inverters, which in turn power the AC vital buses. Additionally, they power the emergency DC lighting system.

The vital DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System - Operating," and LCO 3.8.10, "Distribution Systems - Shutdown."

Each vital battery has adequate storage capacity to carry the required load continuously for at least 4 hours in the event of a loss of all AC power (station blackout) without an accident or for 30 minutes with an accident considering a single failure. Load shedding of nonrequired loads will be performed to achieve the required coping duration for station blackout conditions.

Each 125 VDC vital battery is separately housed in a ventilated room apart from its charger and distribution centers, except for Vital Battery V. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

The batteries for the vital DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles, derated for minimum ambient temperature and the

(continued)

BASES

BACKGROUND

125 V Vital DC Electrical Power Subsystem (continued)

100% design demand. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery (132 V for Vital Battery V). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each Vital DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery bank from the design minimum charge to its fully charged state within 12 hours (with accident loads being supplied) following a 30 minute AC power outage and in approximately 36 hours (while supplying normal steady state loads following a 2 hour AC power outage), (Ref. 6).

125 V Diesel Generator (DG) DC Electrical Power Subsystem

Control power for the DGs is provided by five DG battery systems, one per DG. Each system is comprised of a battery, a dual battery charger assembly, distribution center, cabling, and cable ways. The DG 125V DC control power and field-flash circuits have power supplied from their respective 125V distribution panel. The normal supply of DC current is from the associated charger. The battery provides control and field-flash power when the charger is unavailable. The charger supplies the normal DC loads, maintains the battery in a fully charged condition, and recharges (480V AC available) the battery while supplying the required loads regardless of the status of the unit. The batteries are physically and electrically independent. The battery has sufficient capacity when fully charged to supply required loads for a minimum of 30 minutes following a loss of normal power. Each battery is normally required to supply loads during the time interval between loss of normal feed to its charger and the receipt of emergency power to the charger from its respective DG.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Section 6 (Ref. 7), and in the FSAR, Section 15(Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The vital DC electrical power system provides normal and emergency DC electrical power for the emergency auxiliaries, and control and switching during all power for the emergency auxiliaries, and control and switching during all MODES of operation. The DG battery systems provide DC power for the DGs.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; and

b. A worst case single failure.

The DC sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

Four 125V vital DC electrical power subsystems, each vital subsystem channel consisting of a battery bank, associated battery charger and the corresponding control equipment and interconnecting cabling supplying power to the associated DC bus within the channel; and four DG DC electrical power subsystems each consisting of a battery, a dual battery charger assembly, and the corresponding control equipment and interconnecting cabling are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (A00) or a postulated DBA. Loss of any DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE vital DC electrical power subsystem requires all required batteries and respective chargers to be operating and connected to the associated DC buses.

The LCO is modified by two Notes. Note 1 indicates that Vital Battery V may be substituted for any of the required vital batteries. However, the fifth battery cannot be declared OPERABLE until it is connected electrically in place of another battery and it has satisfied applicable Surveillance Requirements. Note 2 has been added to indicate that the C-S DG and its associated DC subsystem may be substituted for any of the required DGs. However, the C-S DG and its associated DC subsystem cannot be declared OPERABLE until it is connected electrically in place of another DG, and it has satisfied applicable Surveillance Requirements.

(continued)

BASES (continued)

APPLICABILITY

The four vital DC electrical power sources and four DG DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe plant operation and to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOs or abnormal transients; and

b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources - Shutdown."

ACTIONS

A.1

Condition A represents one vital channel with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is, therefore, imperative that the operator's attention focus on stabilizing the plant, minimizing the potential for complete loss of DC power to the affected train. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution system train.

If one of the required vital DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining vital DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure of the OPERABLE subsystem would, however, result in a situation where the ability of the 125V DC electrical power subsystem to support its required ESF function is not assured, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 8) and reflects a reasonable time to assess plant status as a function of the inoperable vital DC electrical power subsystem and, if the vital DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe plant shutdown.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.4.11 (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance may perturb the electrical distribution system and challenge safety systems. This Surveillance is normally performed during MODES 5 and 6 since it would require the DC electrical power subsystem to be inoperable during performance of the test. However, this Surveillance may be performed in MODES 1, 2, 3, or 4 provided the Vital Battery V is substituted in accordance with LCO Note 1. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.4.12

This SR requires that each diesel generator battery charger be capable of recharging its associated battery from a capacity or service discharge test while supplying normal loads, or alternatively, operating at current limit for a minimum of 4 1/2 hours at a nominal 125 VDC. This requirement is based on the design capacity of the chargers (Ref. 13) and their performance characteristic of current limit operation for a substantial portion of the recharge period. Battery charger output current is limited to a maximum of 140% of the 20 amp rated output. Recharging the battery verifies the output capability of the charger can be sustained, that current limit adjustments are properly set and that protective devices will not inhibit performance at current limit settings. According to Regulatory Guide 1.32 (Ref. 6), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the plant during these demand occurrences. Verifying the capability of the charger to operate in a sustained current limit condition ensures that these requirements can be satisfied.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.4.12 (continued)

The Surveillance Frequency is acceptable, given the plant conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

For the DG DC electrical subsystem, this Surveillance may be performed in MODES 1, 2, 3, or 4 in conjunction with LCO 3.8.1.B since the DG DC electrical power subsystem supplies loads only for the inoperable diesel generator and would not otherwise challenge safety systems supplied from vital electrical distribution systems. If available, the C-S DG and its associated DC electrical power subsystem may be substituted in accordance with LCO Note 2. Additionally, credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Unexpected operational events which cause the equipment to perform the function specified by this Surveillance, for which adequate documentation of the required performance is available; and

2) Post corrective maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to maintain OPERABILITY or reliability.

SR 3.8.4.13

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to worst case design duty cycle requirements based on Reference 10 and 12.

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 6) and Regulatory Guide 1.129 (Ref. 11), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests, not to exceed 18 months.

(continued)

BASES

LCO (continued)	CVI signal will initiate that portion of the ABI normally initiated by the spent fuel pool radiation monitors. In addition, the ABGTS must remain operable if these containment penetrations are open to the Auxiliary Building during movement of irradiated fuel in side containment.
APPLICABILITY	An initial assumption in the analysis of a fuel handling accident inside containment is that the accident occurs while irradiated fuel is being handled. Therefore, LCO 3.9.8 is applicable only at this time. See additional discussion in the Applicable Safety Analysis and LCO sections.
ACTIONS	A.1 and A.2 If one Reactor Building Purge Air Cleanup Unit is inoperable, that air cleanup unit must be isolated. This places the system in the required accident configuration, thus allowing refueling to continue after verifying the remaining air cleanup unit is aligned and OPERABLE. The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner. B.1 With two Reactor Building Purge Air Cleanup Units inoperable, movement of irradiated fuel assemblies within containment must be suspended. This precludes the possibility of a fuel handling accident in containment with both Reactor Building Purge Air Cleanup Units inoperable. Performance of this action shall not preclude moving a component to a safe position. The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.
SURVEILLANCE REQUIREMENTS	SR 3.9.8.1 The Ventilation Filter Testing Program (VFTP) encompasses the Reactor Building Purge Air Cleanup Unit filter tests in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.8.1 (continued)

the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. Watts Bar FSAR, Section 15.5.6, "Environmental Consequences of a Postulated Fuel Handling Accident."

2. Watts Bar FSAR, Section 9.4.6, "Reactor Building Purge Ventilating System."

3. Regulatory Guide 1.52 (Rev. 02), "Design, Testing and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."