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

MAR 2 7 2001

10 CFR 50.71(e)

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

Gentlemen:

In the Matter of) Docket No. 50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - CHANGES MADE TO THE WBN TECHNICAL SPECIFICATION BASES AND TECHNICAL REQUIREMENTS MANUAL

The purpose of this letter is to provide the NRC with copies of changes to the WBN Technical Specification Bases (TS Bases), through Revision 42, and WBN Technical Requirements Manual (TRM), through Revision 24, in accordance with WBN TS Section 5.6, "TS Bases Control Program," and WBN TRM Section 5.1, "Technical Requirements Control Program," respectively. These changes have been implemented at WBN during the period since WBN's last update (October 18, 1999) and meet criteria described within the above control programs for which prior NRC approval is not required. Both control programs require such changes to be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). WBN's FSAR update in accordance with 10 CFR 50.71(e) is being provided under separate cover. In addition, this letter provides a replacement table of contents including an effective page listing for the WBN Technical Specifications (TS), the TS Bases, and the TRM.

Enclosure 1 provides the WBN TS Table of Contents and List of Effective Pages. Enclosure 2 provides the Table of Contents and List of Effective Pages for the TS Bases and the changes

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to the TS Bases addressed above. Note that changes made to the TS Bases under approved license amendments to the WBN TS are not included unless necessary for page integrity. The attachment to Enclosure 2 provides changed TS Bases pages made during this reporting period which were superseded by changed pages provided under Enclosure 2. Enclosure 3 provides the Table of Contents and List of Effective Pages for the WBN TRM and the changes to the TRM addressed above. Changes to the TS Bases and TRM made only to address pagination or format are not included in this transmittal.

If you have any questions, please contact me at (423) 365-1824.

Sincerely,

P. L. Pace, Manager
Licensing and Industry Affairs

Enclosure
cc (Enclosure):
NRC Resident Inspector
Watts Bar Nuclear Plant
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Spring City, Tennessee 37381

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LIST OF TABLES

LIST OF FIGURES

Acronym	Title
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
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RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SR	Surveillance Requirement
UHS	Ultimate Heat Sink

TECHNICAL SPECIFICATIONS

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ii	0	Initial
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iv	0	Initial
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* Initial is the effective date of the WBN 1 Full Power License. Initial Issue pages do not have a revision level, date, or amendment number.

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TECHNICAL SPECIFICATIONS

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AMENDMENT LISTING

Amendments	Issued	Subject
NPF-20 Low Power Operating License	11-09-95	
NPF-90 Full Power Operating License	02-07-96	
Amendment 1	02-28-96	Turbine Driven AFW Pump Suction requirement
Amendment 2	06-13-96	Ice Bed Surveillance frequency and weight.
Amendment 3	09-09-96	Ice Condenser Lower Inlet Door Surveillance.
Amendment 4	10-15-96	Operations Manager Requirements.
Amendment 5	05-27-97	Appendix J, Option B.
Amendment 6	07-28-97	Spent Fuel Pool Rerack.
Amendment 7	09-11-97	Cycle 2 Core Reload.
Amendment 8	09-15-97	Tritium Producing Burnable Poison Rods (TPBAR) Lead Test Assembly (LTA).
Amendment 9	10-21-97	PTLR Methodology.
Amendment 10	06-09-98	Hydrogen Mitigation System Temporary Specification.
Amendment 11	08-10-98	Relocation of F(Q) Penalty to COLR
Amendment 12	10-19-98	Online testing of the diesel batteries and performance of the 24 hour diesel endurance run.
Amendment 13	10-26-98	Clarification of Surveillance Testing Requirements for TDAFW Pump
Amendment 14	11-10-98	COMS - Four Hour Allowance to Make RHR Suction Relief Valve Operable after Mode 4 Entry
Amendment 15	12-01-98	Increase of Enrichment Limit to Five Percent for New Fuel Storage

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Amendment 16	12-17-98	New action for Steam Generator ADVs due to Inoperable ACAS.
Amendment 17	12-30-98	Slave Relay Surveillance Extension to 18 Months
Amendment 18	01-15-99	Deletion of Power Range Neutron Flux High Negative Rate Reactor Trip Function
Amendment 19	03-07-00	Reset Power Range High Flux Reactor Trip Setpoints for Multiple Inoperable MSSVs.
Amendment 20	03-14-00	Change to a More Negative Moderator Temperature Coefficient (MTC)
Amendment 21	03-17-00	Best Estimate Large Break Loss-of-Coolant Accident Analysis
Amendment 22	03-17-00	Ice Condenser sampling surveillance requirements, SR 3.6.11.5 and SR 3.6.11.7.
Amendment 23	03-22-00	For SR 3.3.2.10, Table 3.3.2-1, added note h for one time relief from turbine trip response time testing.
Amendment 24	06-13-00	Elimination of Response Time Testing
Amendment 25	07-17-00	Ice Condenser flow channel inspection requirements, SR 3.6.11.4
Amendment 26	08-24-00	Administrative Controls for Open Penetrations During Refueling Operations
Amendment 27	09-08-00	Steam Generator Alternate Repair Criteria F-Star (F*) for Tube Sheet Region
Amendment 28	09-13-00	Physics Tests Exceptions
Amendment 29	12-05-00	Physical Security Plan amendment regarding testing frequency for tamper switch/line supervision alarms
Amendment 30	12-08-00	One Time DG Action Completion Time extension from 72 hours to 10 days for the 1B-B DG generator replacement

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AMENDMENT LISTING

Amendments	Issued	Subject
Amendment 31	01-19-01	Power Uprate from 3411 MWt to 3459 MWt Using Leading Edge Flow Meter (LEFM)

TECHNICAL SPECIFICATIONS

LIST OF APPROVED EXEMPTIONS

1. An exemption to 10 CFR 50 Appendix J, Section III.D.2(b)(ii) was granted with the operating license on November 9, 1995 for performing the overall air lock leak test at pressure P_a before establishing containment integrity if air lock maintenance had been performed that have could affected the air lock sealing capability. Justification for this exemption is in WBN Supplemental Safety Evaluation Report (SSER) 4. This Exemption was deleted with the issuance of Amendment 5 on May 27, 1997.

2. An exemption to 10 CFR 70.24 was previously granted in the SSER 5 for the criticality monitoring requirements in the Special Nuclear Material (SNM) License SNM-1861. The exemption was carried forth into the WBN Unit 1 Operating License on November 9, 1995.

3. An exemption to 10 CFR 73.55(c)(10) for the surface vehicle bomb rule was granted from the implementing schedule of fuel load until February 27, 1996. Justification for this exemption is provided in SSER 15 an granted in the WBN Unit 1 Low Power Operating License on November 9, 1995. Implementation was certified by letter dated February 15, 1996 (T04 960215 292)

4. An exemption to certain requirements in 10 CFR 73.55(d)(5) was granted relating to the returning of picture badges upon exit from the protected areas such that individuals not employed by TVA who are authorized unescorted access into the protected areas can take their badges offsite. Justification for this exemption is discussed in SSER 15 and granted by letter dated December 15, 1994 (A00 941219 003).

5. An exemption to 10 CFR 50, Appendix E was granted such that the State of Tennessee which is within the ingestion exposure pathway emergency planning zone, need not participate in the next full participation exercise. This exemption was granted in the Low Power Operating License dated November 9, 1995. Exemption is no longer valid after the full participation exercise which was held on November 15, 1995.

6. An exemption to 10 CFR 50.60 was granted to use the safety margins in ASME Code Case N-514, "Low Temperature Overpressure Protection," in lieu of the safety margins required by 10 CFR 50, Appendix G to establish fracture toughness requirements for the RCS pressure boundary pressure and temperature limits. This exemption was granted on September 29, 1997 (L44 971006 005).

ENCLOSURE 2

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ESFAS	Engineered Safety Features Actuation System
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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

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE*
i	0	Initial
ii	0	Initial
iii	11	07-28-97
iv	0	Initial
v	0	Initial
vi	0	Initial
vii	0	Initial
viii	42	03-07-01
ix	32	04-13-00
x	40	09-13-00
xi	39	03-17-00
xii	34	07-07-00
xiii	13	09-11-97
xiv	34	07-07-00
xv	29	03-13-00
xvi	42	03-07-01
xvii	39	03-17-00
xviii	29	03-13-00
xix	36	08-23-00
xx	41	01-22-01
xxi	41	01-22-01
xxii	35	08-14-00
xxiii	29	03-13-00
xxiv	37	09-08-00
xxv	11	07-28-97
xxvi	18	09-09-98
xxvii	27	01-15-99
xxviii	37	09-08-00
xxix	42	03-07-01

PAGE	REVISION	DATE
B 2.0-1	0	Initial
B 2.0-2	0	Initial
B 2.0-3	0	Initial
B 2.0-4	13	09-11-97
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B 2.0-6	13	09-11-97
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B 2.0-8	0	Initial
B 2.0-9	0	Initial
B 2.0-10	0	Initial
B 2.0-11	0	Initial
B 2.0-12	0	Initial
B 3.0-1	0	Initial
B 3.0-2	0	Initial
B 3.0-3	0	Initial

* Initial is the effective date of the WBN 1 Full Power License. Initial Issue pages do not have a revision level, date, or amendment number.

LIST OF EFFECTIVE PAGES

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B 3.1-33	0	Initial

LIST OF EFFECTIVE PAGES

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B 3.2-7	0	Initial

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

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B 3.2-28	0	Initial
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B 3.3-22	0	Initial

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

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B 3.3-32	13	09-11-97
B 3.3-33	13	09-11-97
B 3.3-34	13	09-11-97
B 3.3-35	13	09-11-97
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TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

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B 3.3-112	0	Initial

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

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B 3.3-141	0	Initial
B 3.3-142	0	Initial
B 3.3-143	0	Initial
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LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
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B 3.3-161	0	Initial
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B 3.3-167	0	Initial
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B 3.3-170	0	Initial
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B 3.3-172	0	Initial
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B 3.4-18	0	Initial
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B 3.4-23	0	Initial
B 3.4-24	0	Initial
B 3.4-25	29	03-13-00

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

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B 3.4-56	42	03-07-01
B 3.4-57	42	03-07-01
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B 3.4-59	0	Initial
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B 3.4-64	0	Initial
B 3.4-65	0	Initial
B 3.4-66	0	Initial
B 3.4-67	22	11-10-98
B 3.4-68	0	Initial
B 3.4-69	0	Initial
B 3.4-70	0	Initial

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PAGE	REVISION	DATE
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B 3.4-72	7	09/28/96
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B 3.5-16	0	Initial
B 3.5-17	0	Initial

TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

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B 3.5-34	29	03-13-00
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B 3.6-2	10	05-27-97
B 3.6-3	10	05-27-97
B 3.6-4	10	05-27-97
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B 3.6-6	5	07-03-96
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B 3.6-9	0	Initial
B 3.6-10	0	Initial
B 3.6-11	0	Initial
B 3.6-12	10	05-27-97
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B 3.6-24	0	Initial
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B 3.6-26	10	05-27-97
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TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

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B 3.6-82	21	11-30-98
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B 3.6-87	0	Initial
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B 3.6-94	0	Initial
B 3.6-95	0	Initial
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B 3.6-98	29	03-13-00
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B 3.7-14	0	Initial
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B 3.7-16	0	Initial
B 3.7-17	0	Initial
B 3.7-18	0	Initial
B 3.7-19	0	Initial

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PAGE	REVISION	DATE
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B 3.7-76	11	07-28-97
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B 3.8-13	0	Initial
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B 3.8-16	0	Initial
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B 3.8-18	29	03-13-00
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B 3.8-28	19	10-19-98
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B 3.8-38	0	Initial
B 3.8-39	0	Initial
B 3.8-40	0	Initial
B 3.8-41	0	Initial
B 3.8-42	0	Initial
B 3.8-43	0	Initial
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B 3.8-46	0	Initial
B 3.8-47	0	Initial
B 3.8-48	29	03-13-00
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B 3.8-50	0	Initial
B 3.8-51	29	03-13-00
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TECHNICAL SPECIFICATIONS BASES

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REVISION LISTING

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

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REVISION LISTING

REVISIONS	ISSUED	SUBJECT
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
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

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REVISION LISTING

REVISIONS	ISSUED	SUBJECT
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.
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

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REVISIONS	ISSUED	SUBJECT
Revision 38	09-17-00	SR 3.2.1.2 was revised to reflect the area of the core that will be flux mapped.
Revision 39 (Amendments 21 and 28)	09-13-00	Amendment 21 - Implementation of Best Estimate LOCA analysis. Amendment 28 - Revision of LCO 3.1.10, "Physics Tests Exceptions - Mode 2."
Revision 40	09-28-00	Clarifies WBN's compliance with ANSI/ANS-19.6.1 and deletes the detailed descriptions of Physics Tests.
Revision 41 (Amendment 31)	01-22-01	Power Uprate from 3411 MWt to 3459 MWt Using Leading Edge Flow Meter (LEFM)
Revision 42	03-07-01	Clarify Operability Requirements for Pressurizer PORVs

BASES

BACKGROUND (continued)

calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel and burnable absorbers deplete, the RCS boron concentration is adjusted to compensate for the net core reactivity change to maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the initial and reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from initial fuel loading or a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of the NRC Policy Statement.

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of $\pm$ 1% $\Delta k/k$ has been established based on engineering judgment. A 1%$\Delta k/k$ deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that any normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability"; General Design Criterion 28, "Reactivity Limits"; and General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than zero. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOC within the range analyzed in the plant accident analysis. For some core designs, the burnable absorbers may burn out faster than the fuel depletes early in the cycle. This may cause the boron concentration to increase with burnup early in the cycle and the most positive MTC not to occur at BOC, but somewhat later in the cycle. For these core designs, an as-measured criterion is established that is sufficiently less positive than zero to ensure that the MTC remains within the LCO upper limit during the cycle. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR accident and transient analyses.

(continued)

BASES

BACKGROUND (continued)	If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality during non-MSLB events, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity. The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to changes in RCS boron concentration associated with fuel and burnable absorber burnup.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is at the most positive value. Such accidents include the rod withdrawal transient from either zero (Ref. 4) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is at the most negative value. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

MTC values are bounded in initial and reload safety evaluations by assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value in order to confirm reload design predictions.

MTC satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.4 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the initial and reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive at or near BOC; this upper bound must not be exceeded. This maximum upper limit occurs at or near BOC, all rods out (ARO), hot zero power conditions. For some core designs, the burnable absorbers may burn out faster than the fuel depletes early in the cycle. This may cause the boron concentration to increase with burnup early in the cycle and the most positive MTC not to occur at BOC, but somewhat later in the cycle. Therefore, an as-measured criterion is established in the COLR that is sufficiently less positive than zero to ensure that the MTC remains within the LCO upper limit during the cycle. This criterion is not an LCO MTC limit; the COLR prescribes appropriate administrative controls for exceeding this value consistent with SR 3.1.4.1. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value (upper limit) that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

(continued)

BASES (continued)

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled CONTROL ROD assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

If the BOC MTC upper limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its upper limit. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

(continued)

BASES

ACTIONS (continued)

B.1

If the required administrative withdrawal limits at BOC are not established within 24 hours, the unit must be brought to MODE 2 with k_{eff} < 1.0 to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, 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 4 within 12 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 3.1.4.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the upper limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after initial fuel loading and refueling. The ARO value can be directly compared to the BOC as-measured criterion provided in the COLR. Compliance with this as-measured criterion ensures that the MTC will remain within the LCO upper limit during the cycle. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 PHYSICS TESTS Exceptions - MODE 1

BASES

BACKGROUND

The primary purpose of the MODE 1 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow the performance of instrumentation calibration tests and special PHYSICS TESTS. The exceptions to LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)" are most often appropriate for xenon stability tests. The exceptions to LCO 3.1.5, "Rod Group Alignment Limits"; LCO 3.1.6, "Shutdown Bank Insertion Limit"; and LCO 3.1.7, "Control Bank Insertion Limits," may be required in the event that it is necessary or desirable to do special PHYSICS TESTS involving abnormal rod or bank configurations.

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 at 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, low power, power

(continued)

BASES

BACKGROUND (continued)

ascension, and at power operation. 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 permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

Typical PHYSICS TESTS for reload fuel cycles (Ref. 4) in MODE 1 are listed below:

a. Neutron Flux Symmetry;

b. Power Distribution - Intermediate Power;

c. Power Distribution - Full Power; and

d. Critical Boron Concentration - Full Power.

The first test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance. The Power Distribution - Intermediate Power test can be performed at a stable power level between 40 and 80 percent RTP. The last two tests are performed at $\geq$ 90% RTP.

APPLICABLE SAFETY ANALYSES

The fuel is protected by an LCO, which preserves the initial conditions of the core assumed during the safety analyses. The methods for development of the LCO, which are superseded by this LCO, are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating controls or process variables to deviate from their LCO limitations.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reference 6 defines requirements for initial testing of the facility, including PHYSICS TESTS. Table 14.2-2 (Ref. 6) summarizes the zero, low power, and power tests. Summaries of typical reload fuel cycle PHYSICS TESTS are available in ANSI/ANS-19.6.1 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in:

LCO 3.1.5, "Rod Group Alignment Limits";
LCO 3.1.6, "Shutdown Bank Insertion Limits";
LCO 3.1.7, "Control Bank Insertion Limits";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; or
LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the requirements of LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)," and LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," are satisfied, power level is maintained $\leq$ 85% RTP, and SDM is $\geq$ 1.6% $\Delta k/k$. Therefore, LCO 3.1.9 requires surveillance of the hot channel factors and SDM to verify that their limits are not being exceeded.

PHYSICS TESTS include measurements of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the component and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

Reference 7 allows special test exceptions to be included as part of the LCO that they affect. However, it was decided to retain this special test exception as a separate LCO because it was less cumbersome and provided additional clarity.

(continued)

BASES (continued)

LCO

This LCO allows selected control rods and shutdown rods to be positioned outside their specified alignment limits and insertion limits to conduct PHYSICS TESTS in MODE 1, to verify certain core physics parameters. The power level is limited to ≤ 85% RTP and the power range neutron flux trip setpoint is set at 10% RTP above the PHYSICS TESTS power level with a maximum setting of 90% RTP. Violation of LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, LCO 3.2.3, or LCO 3.2.4, during the performance of PHYSICS TESTS does not pose any threat to the integrity of the fuel as long as the requirements of LCO 3.2.1 and LCO 3.2.2 are satisfied and provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Power Range Neutron Flux - High trip setpoints are ≤ 10% RTP above the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; and

c. SDM is ≥ 1.6% $\Delta k/k$.

Operation with THERMAL POWER ≤ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. The Power Range Neutron Flux - High trip setpoint is reduced so that a similar margin exists between the steady state condition and the trip setpoint that exists during normal operation at RTP.

APPLICABILITY

This LCO is applicable in MODE 1 when performing PHYSICS TESTS. The applicable PHYSICS TESTS are performed at ≤ 85% RTP. Other PHYSICS TESTS are performed at full power but do not require violation of any existing LCO, and therefore do not require a PHYSICS TESTS exception. The PHYSICS TESTS performed in MODE 2 are covered by LCO 3.1.10, "PHYSICS TESTS Exceptions - MODE 2."

(continued)

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 run an incore flux map 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. Design isothermal temperature coefficient (ITC).

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.

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

(continued)

BASES

BACKGROUND (continued)

all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

Typical PHYSICS TESTS for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration - Control Rods Withdrawn;

b. Control Rod Bank Worth;

c. Isothermal Temperature Coefficient (ITC); and

d. Neutron Flux Symmetry.

The last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Table 14.2-2 summarizes the zero, low power, and power tests for the initial plant startup. Summaries of typical reload fuel cycle PHYSICS TESTS are available in ANSI/ANS-19.6.1 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the requirements specified in LCO 3.1.4, "Moderator Temperature Coefficient MTC)," LCO 3.1.5, LCO 3.1.6, (LCO 3.1.7, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 541°F, and SDM is ≥ 1.6% Δk/k.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One Power Range Neutron Flux channel may be bypassed, reducing the number of required channels from "4" to "3". Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 3, 6, and 16.e, may be reduced to "3" required channels during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 541°F; and

b. SDM is ≥ 1.6% Δ k/k.

(continued)

BASES (continued)

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions - MODE 1."

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

When THERMAL POWER is >5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest T_{avg} is < 541°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 541°F could violate the assumptions for accidents analyzed in the safety analyses.

(continued)

BASES

ACTIONS (continued)

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 an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS.

SR 3.1.10.2

Verification that the RCS lowest loop T_{avg} is $\geq$ 541°F (value does not account for instrument error, Ref. 7) will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.10.3

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

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

e. Xenon concentration;

f. Samarium concentration; and

g. Design isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this 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 occurring 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. WCAP-11618, "MERITS Program - Phase II, Task 5, Criteria Application," dated November 1987, including Addendum 1, April 1989.

7. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."

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

The limit with which $F^C_Q(Z)$ 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 $F^C_Q(Z)$ 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 $F^C_Q(Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F^C_Q(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits).

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, $F^C_Q(Z)$, by W(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F^W_Q(Z)$.

The limit with which $F^W_Q(Z)$ 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. $F^W_Q(Z)$ 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 $F^W_Q(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to $F^M_Q(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

(continued)

BASES (continued)

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

(continued)

BASES

ACTIONS (continued)

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 *

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for verifying that the pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2 *

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for verifying RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3 *

The 12 hour Surveillance Frequency to verify the RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.4 *

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

*Note: The accuracy of the instruments used for monitoring RCS pressure, temperature and flow rate is discussed in this Bases section under LCO (Ref. 2).

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis," Section 15.2, "Normal Operation and Anticipated Transients," and Section 15.3.4, "Complete Loss Of Forced Reactor Coolant Flow."

2. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.4, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly(RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 557°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement.

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2, with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.10, "MODE 2 PHYSICS TESTS Exceptions," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

(continued)

BASES (continued)

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 551°F (value does not account for instrument error, Ref. 2) every 30 minutes when the T_{avg} - T_{ref} deviation alarm is not reset and any RCS loop T_{avg} < 561°F.

The Note modifies the SR. When any RCS loop average temperature is < 561°F and the T_{avg} - T_{ref} deviation alarm is alarming, RCS loop average temperatures could fall below the LCO requirement without additional warning. The SR to verify RCS loop average temperatures every 30 minutes is frequent enough to prevent the inadvertent violation of the LCO.

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis."
2. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."

BASES

ACTIONS (continued)

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

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

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 6% (value does not account for instrument error, Ref. 1) for required RCS loops. If the SG secondary side narrow range water level is < 6%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	<u>SR 3.4.5.3</u> Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.
REFERENCES	1. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

BASES

ACTIONS (continued)

D.1, D.2 and D.3

If no loop is OPERABLE or in operation, all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets. All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. Opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

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

SR 3.4.6.2

This SR requires verification every 12 hours that one RCS or RHR loop is in operation when the rod control system is not capable of rod withdrawal. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.3

SR 3.4.6.3 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 6% (value does not account for instrument error, Ref. 1). If the SG secondary side narrow range water level is < 6%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.4

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

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

BASES

LCO (continued)

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 6% narrow range.

Operation in other MODES is covered by:

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

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels < 6% narrow range redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

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

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are ≥ 6% (value does not account for instrument error, Ref. 1) narrow range ensures an alternate decay heat removal method in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side water level is ≥ 6% narrow range in at least two SGs,

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.7.3 (continued) this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.
REFERENCES	1. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops – MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

(continued)

BASES (continued)

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level - High Trip.

If the pressurizer water level is not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the plant must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the plant out of the applicable MODES and restores the plant to operation within the bounds of the safety analyses.

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

(continued)

BASES

ACTIONS (continued)

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period.

C.1 and C.2

If one group of pressurizer heaters are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper level limit of ≤92% (value does not account for instrument error, Ref. 3) to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

(continued)

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analyses."
2. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
3. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, MODE 5, and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

The upper and lower pressure limits are based on a ± 3% tolerance. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are pilot-operated solenoid valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains (Ref. 1).

The plant has two PORVs, each having a relief capacity of 210,000 lb/hr at 2485 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure - High reactor trip setpoint following a step reduction of 50% of full load with steam dump. In addition,

(continued)

BACKGROUND (continued)

the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR), pressurizer filling, or reactor coolant saturation criteria are critical (Ref. 2). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an

(continued)

LCO (continued)

inoperable PORV that is capable of being manually cycled (e.g., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODES 4, 5, and 6 with the reactor vessel head in place when both pressure and core energy are decreased and the pressure surges become much less significant. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status.

(continued)

ACTIONS (continued)

A.1

PORVs may be inoperable and capable of being manually cycled (e.g., due to excessive seat leakage). In this condition, either the PORV must be restored or the flow path isolated within 1 hour. The associated block valve is required to be closed, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition.

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable and not capable of being manually cycled, it must be either restored, or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE

(continued)

ACTIONS

C.1 and C.2 (continued)

status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an event if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

E.1, E.2, E.3, and E.4

If both PORVs are inoperable and not capable of being manually cycled, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time

(continued)

ACTIONS

E.1, E.2, E.3, and E.4 (continued)

to correct the situation. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

F.1 and F.2

If both block valves are inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

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

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

SR 3.4.11.2

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

REFERENCES

1. Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1977.
2. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency."
3. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Cold Overpressure Mitigation System (COMS)

BASES

BACKGROUND

The COMS controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the COMS MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all safety injection pumps and all but one charging pump incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

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

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

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

SR 3.5.1.2 and SR 3.5.1.3

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

Note:
In the discussion contained in the Applicable Safety Analyses of this Bases section, the borated water volume and nitrogen cover pressure specified for SR 3.5.1.2 and SR 3.5.1.3 account for instrument accuracy (Ref. 6).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

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

SR 3.5.1.5

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

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

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

(continued)

BASES (continued)

REFERENCES

1. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

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

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

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

5. NUREG-1366, Improvements to Technical Specifications Surveillance Requirements, December 1992.

6. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

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

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

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

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 9 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. The specified temperature range is ≥ 60°F and ≤ 105°F and does not account for instrument error (Ref. 2). The 24 hour Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.

SR 3.5.4.2

The required minimum RWST water level is ≥370,000 gallons (value does not account for instrument error, Ref. 2). Verification every 7 days of the presence of this water volume ensures that a sufficient initial supply of water is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.5.4.3 (continued) concentration is appropriate and has been shown to be acceptable through operating experience.
REFERENCES	1. Watts Bar FSAR, Section 6.3, "Emergency Core Cooling System," and Section 15.0, "Accident Analysis." 2. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."

BASES

APPLICABILITY (continued)

is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in these MODES. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTIONS

A.1

With the seal injection flow exceeding its limit, the amount of charging flow available to the RCS may be reduced. Under this Condition, action must be taken to restore the flow to below its limit. The operator has 4 hours from the time the flow is known to be above the limit to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limit. This time is conservative with respect to the Completion Times of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.5.1

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow within the limit listed below ensures that proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained:

> ≤ 40 gpm with charging pump discharge header pressure ≥ 2430 psig and the pressurizer level control valve full open (values do not account for instrument error, Ref. 3).

The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.

As noted, the Surveillance is not required to be performed until 4 hours after the RCS pressure has stabilized within a ± 20 psig range of normal operating pressure. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

REFERENCES

1. Watts Bar FSAR, Section 6.3, "Emergency Core Cooling System," and Section 15.0, "Accident Analysis."

2. Title 10, Code of Federal Regulations, Part 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Plants," 1974.

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment was also designed for an external pressure load equivalent to 2.0 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was -0.1 psig. This resulted in a minimum pressure inside containment of 1.4 psig, which is less than the design load.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of the NRC Policy Statement.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System or Air Return Fans.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODES 5 or 6.

(continued)

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

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

REFERENCES

1. Watts Bar FSAR, Section 6.2.1, "Containment Functional Design."
2. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models."
3. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

BASES

APPLICABLE SAFETY ANALYSES (continued)	Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.
LCO	During a DBA, with an initial containment average air temperature within the LCO temperature limits, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured. In MODES 2, 3 and 4, containment air temperature may be as low as 60°F (value does not account for instrument error) because the resultant calculated peak containment accident pressure would not exceed the design pressure due to a lesser amount of energy released from the pipe break in these MODES (Ref. 4).
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature in the upper or lower compartment is not within the limit of the LCO, the average air temperature in the affected compartment must be restored to within limits within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1 and SR 3.6.5.2

LCO 3.6.5 specifies that the containment average air temperature shall be the following values which do not account for instrument error (Ref. 3):

a. ≥ 85°F and ≤ 110°F for the containment upper compartment, and

b. ≥ 100°F and ≤ 120°F for the containment lower compartment.

Verifying that containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, a weighted average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of these SRs is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

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

2. Watts Bar System Description N3-30RB-4002R5, "Reactor Building Ventilation System."

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

4. Westinghouse Letter WAT-D-10698, dated November 23, 1999.

BASES

APPLICABLE SAFETY ANALYSES (continued)

within 18 seconds (20 seconds from the initiating event.) This does not include 10 seconds for diesel generator startup. The analysis shows that the annulus pressure will rise to a value above the EGTS negative pressure control setpoint (become less negative) but will not go positive.

The EGTS satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one EGTS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the EGTS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the Filtration System is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.9.1

Operating each EGTS train for ≥ 10 hours ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours eliminates moisture on the adsorbers and HEPA filters. Experience from filter testing at operating units indicates that the 10 hour period is adequate for moisture elimination on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls, the two train redundancy available.

SR 3.6.9.2

This SR verifies that the required EGTS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP-Technical Specification Section 5.7.2.14). The EGTS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP. It should be noted that for the EGTS, the VFTP pressure drop value across the entire filtration unit does not account for instrument error (Ref. 5).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.9.3

The automatic startup ensures that each EGTS train responds properly. This testing includes the automatic swapping logic of the EGTS pressure control isolation valves in response to the actuation signal. Performance of this swapping logic test will ensure the availability of EGTS functions in the event of an initial single failure of one of the pressure control loops. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the SR interval was developed considering that the EGTS equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.9.1.

SR 3.6.9.4

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

REFERENCES

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.10 Air Return System (ARS)

BASES

BACKGROUND

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

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

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

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

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

(continued)

BASES (continued)

ACTIONS

A.1

If the ice bed is inoperable, it must be restored to OPERABLE status within 48 hours. The Completion Time was developed based on operating experience, which confirms that due to the very large mass of stored ice, the parameters comprising OPERABILITY do not change appreciably in this time period. Because of this fact, the Surveillance Frequencies are long (months), except for the ice bed temperature, which is checked every 12 hours. If a degraded condition is identified, even for temperature, with such a large mass of ice it is not possible for the degraded condition to significantly degrade further in a 48 hour period. Therefore, 48 hours is a reasonable amount of time to correct a degraded condition before initiating a shutdown.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.11.1

Verifying that the maximum temperature of the ice bed is ≤ 27°F (value does not account for instrument error, Ref. 3) ensures that the ice is kept well below the melting point. The 12 hour Frequency was based on operating experience, which confirmed that, due to the large mass of stored ice, it is not possible for the ice bed temperature to degrade significantly within a 12 hour period and was also based on assessing the proximity of the LCO limit to the melting temperature.

Furthermore, the 12 hour Frequency is considered adequate in view of indications in the control room, including the alarm, to alert the operator to an abnormal ice bed temperature condition. This SR may be satisfied by use of the Ice Bed Temperature Monitoring System.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.2

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

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

If a basket is found to contain < 1236 lb of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The average weight of ice in these 21 baskets (the discrepant basket and the 20 additional baskets) shall be ≥ 1236 lb at a 95% confidence level.

Weighing 20 additional baskets from the same bay in the event a Surveillance reveals that a single basket contains < 1236 lb ensures that no local zone exists that is grossly deficient in ice. Such a zone could experience early melt out during a DBA transient, creating a path for steam to pass through the ice bed without being condensed. The Frequency of 18 months was based on ice storage tests and the allowance built into the required ice mass over and above the mass assumed in the safety analyses. Operating experience has verified that, with the 18 month Frequency, the weight requirements are maintained with no significant degradation between surveillances.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.6

This SR ensures that a representative sampling of ice baskets, which are relatively thin walled, perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. Each ice basket must be raised at least 10 feet for this inspection. However, for baskets where vertical lifting height is restricted due to overhead obstruction, a camera shall be used to perform the inspection. The Frequency of 40 months for a visual inspection of the structural soundness of the ice baskets is based on engineering judgment and considers such factors as the thickness of the basket walls relative to corrosion rates expected in their service environment and the results of the long term ice storage testing.

SR 3.6.11.7

This SR ensures that initial ice fill and any subsequent ice additions meet the boron concentration and pH requirements of SR 3.6.11.5. The SR is modified by a NOTE that allows the chemical analysis to be performed on either the liquid or resulting ice of each sodium tetraborate solution prepared. If ice is obtained from offsite sources, then chemical analysis data must be obtained for the ice supplied.

REFERENCES

1. Watts Bar FSAR, Section 6.2, "Containment Systems"
2. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models"
3. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."
4. Westinghouse Letter, WAT-D-10686, "Upper Limit Ice Boron Concentration In Safety Analysis"

B 3.6 CONTAINMENT SYSTEMS

B 3.6.12 Ice Condenser Doors

BASES

BACKGROUND

The ice condenser doors consist of the inlet doors, the intermediate deck doors, and the top deck doors. The functions of the doors are to:

a. Seal the ice condenser from air leakage during the lifetime of the plant; and

b. Open in the event of a Design Basis Accident (DBA) to direct the hot steam air mixture from the DBA into the ice bed, where the ice would absorb energy and limit containment peak pressure and temperature during the accident transient.

Limiting the pressure and temperature following a DBA reduces the release of fission product radioactivity from containment to the environment.

The ice condenser is an annular compartment enclosing approximately 300°of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The inlet doors separate the atmosphere of the lower compartment from the ice bed inside the ice condenser. The top deck doors are above the ice bed and exposed to the atmosphere of the upper compartment. The intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. This plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets held in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open,

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.15.1

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

SR 3.6.15.2

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

SR 3.6.15.3

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

SR 3.6.15.4

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

(continued)

BASES (continued)

REFERENCES

1. Regulatory Guide 1.52, Revision 2, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants."
2. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains a volume of ≥ 200,000 gallons (value accounts for instrument error, Ref. 5) of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. Watts Bar FSAR, Section 9.2.6, "Condensate Storage Facilities."
2. Watts Bar FSAR, Chapter 6, "Engineered Safety Features."
3. Watts Bar FSAR, Chapter 15, "Accident Analyses."
4. TVA Calculation HCG-LCS-043085, "Minimum CST Water Level Required to Support the AFW System."
5. Watts Bar Drawing 1-47W605-242, "Electrical Tech Spec Compliance Tables."

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling System (CCS)

BASES

BACKGROUND

The CCS provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCS also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCS serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Essential Raw Cooling Water (ERCW) System, and thus to the environment.

The CCS is arranged as two independent, full-capacity cooling trains, Train A and B. Train A in unit 1 is served by CCS Hx A and CCS pump 1A-A. Pump 1B-B, which is actually Train B equipment, is also normally aligned to the Train A header in unit 1. However, pump 1B-B can be realigned to Train B on loss of Train A.

Train B is served by CCS Hx C. Normally, only CCS pump C-S is aligned to the Train B header since few nonessential, normally-operating loads are assigned to Train B. However, pump 1B-B can be realigned to the Train B header on a loss of the C-S pump.

Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential components will be manually isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the CCS is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This may be during a normal or post accident cooldown and shutdown.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

requires a 30 day supply of cooling water in the UHS.

The UHS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains water at or below the maximum temperature that would allow the ERCW System to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the ERCW System. To meet this condition, the UHS temperature should not exceed 85°F.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1

If the UHS is inoperable, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in 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.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies that the ERCW System is available to cool the CCS to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.9.1 (continued) is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is ≤ 85°F (value does not account for instrument error, Ref. 3).
REFERENCES	1. Watts Bar FSAR, Section 9.2.5, "Ultimate Heat Sink." 2. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Revision 1, March 1974. 3. Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."

BASES

ACTIONS

C.1 and C.2 (continued)

handling accident. This does not preclude the movement of fuel assemblies to a safe position.

D.1

When two trains of the ABGTS are inoperable during movement of irradiated fuel assemblies in the fuel handling area, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

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

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

SR 3.7.12.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 8.

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 and -0.5 inches water gauge (value does not account for instrument error, Ref. 10) with respect to atmospheric pressure at a nominal flow rate ≥ 9300 and ≤ 9900 cfm. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 9).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 8.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, "Engineered Safety Feature (ESF) Filter Systems."
2. Watts Bar FSAR, Section 9.4.2, "Fuel Handling Area Ventilation System."
3. Watts Bar FSAR, Section 15.0, "Accident Analysis."
4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."

(continued)

REFERENCES (continued)

5. Regulatory Guide 1.25, March 1972, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

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

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

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

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

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

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the fuel storage pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section 15.4.5 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4).The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits. According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop. The fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

The 10 second start requirement is not applicable to SR 3.8.1.2 (see Note 2) when a modified start procedure as described above is used. If a modified start is not used, the 10 second start requirement of SR 3.8.1.7 applies. Stable operation at the nominal voltage and frequency values is also essential to establishing DG OPERABILITY, but a time constraint is not imposed. This is because a typical DG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not dampened out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the SR. In lieu of a time constraint in the SR, WBN will monitor and trend the actual time to reach steady state operation as a means of ensuring there is no voltage regulator or governor degradation which could cause a DG to become inoperable.

Since SR 3.8.1.7 requires a 10 second start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

The normal 31 day Frequency for SR 3.8.1.2 (see Table 3.8.1-1, "Diesel Generator Test Schedule," in the accompanying LCO) is consistent with Regulatory Guide 1.9 (Ref. 3). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.3 (continued)

rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The 31 day Frequency for this Surveillance (Table 3.8.1-1) is consistent with Regulatory Guide 1.9 (Ref. 3).

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR.

A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in each DG skid mounted day tank is at or above the level (≥ 218.5 gallons, value does not account for instrument error, Ref. 10) at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of DG operation at full load plus 10%.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are provided and facility operators would be aware of any large uses of fuel oil during this period.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Diesel Generator Test Schedule

The DG test schedule (Table 3.8.1-1) implements the recommendations of Revision 3 to Regulatory Guide 1.9 (Ref. 3). The purpose of this test schedule is to provide timely test data to establish a confidence level associated with the goal to maintain DG reliability > 0.975 per demand.

According to Regulatory Guide 1.9, Revision 3 (Ref. 3), each DG should be tested at least once every 31 days. Whenever a DG has experienced 4 or more valid failures in the last 25 valid tests, the maximum time between tests is reduced to 7 days. Four failures in 25 valid tests is a failure rate of 0.16, or the threshold of acceptable DG performance, and hence may be an early indication of the degradation of DG reliability. When considered in the light of a long history of tests, however, 4 failures in the last 25 valid tests may only be a statistically probable distribution of random events. Increasing the test Frequency will allow for a more timely accumulation of additional test data upon which to base judgment of the reliability of the DG. The increased test Frequency must be maintained until seven consecutive, failure free tests have been performed.

The Frequency for accelerated testing is 7 days, but no less than 24 hours. Tests conducted at intervals of less than 24 hours may be credited for compliance with Required Actions. However, for the purpose of re-establishing the normal 31-day Frequency, a successful test at an interval of less than 24 hours should be considered an invalid test and not count towards the 7 consecutive failure free starts, and the consecutive test count is not reset.

A test interval in excess of 7 days (or 31 days as appropriate) constitutes a failure to meet the SRs and results in the associated DG being declared inoperable. It does not, however, constitute a valid test or failure of the DG, and any consecutive test count is not reset.

(continued)

BASES (continued)

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix A, General Design Criterion (GDC) 17, "Electrical Power Systems."

2. Watts Bar FSAR, Section 8.2, "Offsite Power System," and Tables 8.3-1 to 8.3-3, "Safety-Related Standby Power Sources and Distribution Boards," "Shutdown Board Loads Automatically Tripped Following a Loss of Nuclear Unit and Preferred Power," and "Diesel Generator Load Sequentially Applied Following a Loss of Nuclear Unit and Preferred Power."

3. Regulatory Guide 1.9, Rev. 3, "Selection, Design, Qualification and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," July 1993.

4. Watts Bar FSAR Section 6, "Engineered Safety Features."

5. Watts Bar FSAR, Section 15.4, "Condition IV-Limiting Faults."

6. Regulatory Guide 1.93, Rev. 0, "Availability of Electric Power Sources," December 1974.

7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.

8. Title 10, Code of Federal Regulations, Part 50, Appendix A, GDC 18, "Inspection and Testing of Electric Power Systems."

9. Regulatory Guide 1.137, Rev. 1, "Fuel Oil Systems for Standby Diesel Generators," October 1979.

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

BASES

ACTIONS (continued)

B.1

With lube oil inventory < 287 gal per diesel engine, sufficient lubricating oil to support 7 days of continuous DG operation at full load conditions may not be available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.5. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, resampling and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.4 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This

(continued)

BASES

ACTIONS

D.1 (continued)

restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

With starting air receiver pressure < 190 psig, sufficient capacity for five successive DG start attempts does not exist. However, as long as the receiver pressure is ≥ 170 psig (value does not account for instrument error, Ref. 7), there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit of ≥ 190 psig (value does not account for instrument error, Ref. 7). A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil, lube oil or starting air subsystem not within limits for reasons other than addressed by Conditions A through D, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory (≥ 56,754 gallons, value does not account for instrument error, Ref. 7) of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The 7 day period is sufficient time to place the plant in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

(continued)

BASES

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 verifies by visual inspection, that the exposed fuel oil system piping is free of leaks. This test is performed while the DG is running to provide adequate assurance of piping leak tightness and weld integrity. The 18 month Frequency is based on engineering judgement 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 Containment in Aviation Turbine Fuels."
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, six battery chargers (including two 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)

BASES

ACTIONS

D.1 and D.2 (continued)

at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

With two trains with one or more inoperable distribution subsystems that result in a loss of safety function, adequate core cooling, containment OPERABILITY, and other vital functions for DBA mitigation would be compromised, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required AC, vital DC, and AC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical trains is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, vital DC, and AC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. Watts Bar FSAR, Section 6 "Engineering Safety Features," Section 8 "Electric Power," and Section 15 "Accident Analysis."
2. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

TYPE	VOLTAGE	TRAIN A*	TRAIN B*
AC safety buses	6900 V 480 V	Shdn Bd 1A-A, 2A-A Shdn Bd 1A1-A, 1A2-A 2A1-A, 2A2-A Rx MOV Bd 1A1-A, 1A2-A 2A1-A,** 2A2-A C & A Vent Bd 1A1-A, 1A2-A 2A1-A, 2A2-A Diesel Aux Bd 1A1-A, 1A2-A 2A1-A, 2A2-A Rx Vent Bd 1A-A, 2A-A**	Shdn Bd 1B-B, 2B-B Shdn Bd 1B1-B, 1B2-B 2B1-B, 2B2-B Rx MOV Bd 1B1-B, 1B2-B 2B1-B,** 2B2-B C & A Vent Bd 1B1-B, 1B2-B 2B1-B, 2B2-B Diesel Aux Bd 1B1-B, 1B2-B 2B1-B, 2B2-B Rx Vent Bd 1B-B, 2B-B**
AC vital buses	120 V	Vital channel 1-I Vital channel 2-I Vital channel 1-III Vital channel 2-III	Vital channel 1-II Vital channel 2-II Vital channel 1-IV Vital channel 2-IV
DC buses	125 V	Board I Board III	Board II Board IV

* Each train of the AC and DC electrical power distribution systems is a subsystem.

** For Unit 1, 480V Reactor MOV Boards 2A1-1 and 2B1-1 and 480V Reactor Vent Boards 2A-A and 2B-B are available for economic and operational convenience. The boards are considered part of the Unit 1 Electrical Power Distribution System and meet Unit 1 T/S Requirements and testing only while connected. WBN Unit 1 is designed to be operated, shutdown, and maintained in a safe shutdown status without any of these boards or their loads. As such, the boards may be disconnected from service without entering an LCO provided their loads are not substituting for a T/S required load.

ATTACHMENT TO ENCLOSURE 2

TS BASES CHANGED PAGES - SUPERSEDED

BASES

ACTIONS (continued)

C.1

When the RCS lowest T_{avg} is < 541°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 541°F could violate the assumptions for accidents analyzed in the safety analyses.

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 an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.10.2

Verification that the RCS lowest loop T_{avg} is ≥ 541°F (value does not account for instrument error, Ref. 7) will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.10.3

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

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Design isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this 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 occurring 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.

(continued)

BASES

REFERENCES (continued)	4.	ANSI/ANS-19.6.1-1985, "Reload Startup Physics Tests for Pressurized Water Reactors," American National Standards Institute, December 13, 1985.
	5.	WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
	6.	WCAP-11618, "MERITS Program - Phase II, Task 5, Criteria Application," dated November 1987, including Addendum 1, April 1989.
	7.	Watts Bar Drawing 1-47W605-243, "Electrical Tech Spec Compliance Tables."

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.6

This SR ensures that a representative sampling of ice baskets, which are relatively thin walled, perforated cylinders, have not been degraded by wear, cracks, corrosion, or other damage. Each ice basket must be raised at least 10 feet for this inspection. However, for baskets where vertical lifting height is restricted due to overhead obstruction, a camera shall be used to perform the inspection. The Frequency of 40 months for a visual inspection of the structural soundness of the ice baskets is based on engineering judgment and considers such factors as the thickness of the basket walls relative to corrosion rates expected in their service environment and the results of the long term ice storage testing.

REFERENCES

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

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

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.12 Ice Condenser Doors

BASES

BACKGROUND

The ice condenser doors consist of the inlet doors. the intermediate deck doors. and the top deck doors. The functions of the doors are to:

a. Seal the ice condenser from air leakage during the lifetime of the plant: and

b. Open in the event of a Design Basis Accident (DBA) to direct the hot steam air mixture from the DBA into the ice bed. where the ice would absorb energy and limit containment peak pressure and temperature during the accident transient.

Limiting the pressure and temperature following a DBA reduces the release of fission product radioactivity from containment to the environment.

The ice condenser is an annular compartment enclosing approximately 300°of the perimeter of the upper containment compartment. but penetrating the operating deck so that a portion extends into the lower containment compartment. The inlet doors separate the atmosphere of the lower compartment from the ice bed inside the ice condenser. The top deck doors are above the ice bed and exposed to the atmosphere of the upper compartment. The intermediate deck doors. located below the top deck doors. form the floor of a plenum at the upper part of the ice condenser. This plenum area is used to facilitate surveillance and maintenance of the ice bed.

The ice baskets held in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a DBA.

In the event of a DBA. the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

handling accident. This does not preclude the movement of fuel assemblies to a safe position.

D.1

When two trains of the ABGTS are inoperable during movement of irradiated fuel assemblies in the fuel handling area, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

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

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

SR 3.7.12.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.3

This SR verifies that each ABGTS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with Reference 8.

SR 3.7.12.4

This SR verifies the integrity of the ABSCE. The ability of the ABSCE to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the ABGTS. During the post accident mode of operation, the ABGTS is designed to maintain a slight negative pressure in the ABSCE, to prevent unfiltered LEAKAGE. The ABGTS is designed to maintain a negative pressure between -0.25 and -0.5 inches water gauge (value does not account for instrument error, Ref. 10) with respect to atmospheric pressure at a nominal flow rate ≥ 9300 and ≤ 9900 cfm. Periodic testing of ABGTS shall be performed once with one AB general supply fan running and one train of its discharge ABSCE isolation dampers open and once with one purge air supply fan running and one of its suction-side ABSCE isolation dampers open. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 9).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 8.

REFERENCES

1. Watts Bar FSAR, Section 6.5.1, "Engineered Safety Feature (ESF) Filter Systems."

2. Watts Bar FSAR, Section 9.4.2, "Fuel Handling Area Ventilation System."

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

4. Watts Bar FSAR, Section 6.2.3, "Secondary Containment Functional Design."

(continued)

ENCLOSURE 3

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WBN TECHNICAL REQUIREMENTS MANUAL

TECHNICAL REQUIREMENTS

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LIST OF MISCELLANEOUS REPORTS AND PROGRAMS

LIST OF ACRONYMS

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

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1.0 USE AND APPLICATION

1.1 Definitions

-----------------------------------NOTE-----------------------------------
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Requirements and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Requirement that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input.. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

(continued)

1.1 Definitions (continued)

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or other reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

(continued)

1.1 Definitions

LEAKAGE (continued)

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) tube to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG tube LEAKAGE) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

1.1 Definitions (continued)

QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
TSR 3.3.3.1	Verify the Movable Incore Detection System is OPERABLE by: a) Determining each detector's operating voltage; and b) Performing a drift check on each detector.	24 hours

TR 3.3 INSTRUMENTATION

TR 3.3.4 Seismic Instrumentation

TR 3.3.4 The seismic monitoring instrumentation shown in Table 3.3.4-1 shall be OPERABLE.

APPLICABILITY: At all times.

-----------------------NOTE---------------------------
TR 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more seismic monitoring instruments in Panel 0-R-113 or foundation instrument 0-XT-52-75A in the Containment annulus inoperable for > 30 days. OR One or more remaining seismic monitoring instruments inoperable for >60 days.	A.1 Prepare and submit a report to the Commission in accordance with 10 CFR 50.4 outlining the cause of the malfunction and the plans for restoring the instruments(s) to OPERABLE status.	10 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------NOTE--------- All Required Actions must be completed whenever this Condition is entered. -------------------- One or more seismic monitoring instruments actuated during a seismic event.	B.1	Analyze data retrieved from 0-XT-52-75A to determine the magnitude of the vibratory ground motion.	4 hours
	AND		
	B.2	If OBE exceedance is verified, perform walkdowns of key plant equipment and structures to determine extent of damage.	8 hours
	AND		
	B.3	Restore each actuated monitoring instrument to OPERABLE status.	24 hours
	AND		
	B.4	Perform a CHANNEL CALIBRATION on each actuated monitoring instrument.	10 days
	AND		
	B.5	Analyze data retrieved from remaining seismic monitoring instruments.	14 days
	AND		
	B.6	Prepare a report to the NRC in accordance with 10 CFR 50.4 describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.	14 days

TECHNICAL SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
Refer to Table 3.3.4-1 to determine which Technical Surveillance Requirements apply for each seismic monitoring instrument.

	SURVEILLANCE	FREQUENCY
TSR 3.3.4.1	Perform CHANNEL CHECK.	31 days
TSR 3.3.4.2	Perform CHANNEL OPERATIONAL TEST.	184 days
TSR 3.3.4.3	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.4-1 (Page 1 of 1)

Seismic Monitoring Instrumentation

INSTRUMENTS AND SENSOR LOCATIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	MEASUREMENT RANGE
1. Strong Motion Triaxial Accelerometers(1) (5)			
a. 0-XT-52-75A Annulus El. 703	1	TSR 3.3.4.1(2) TSR 3.3.4.2(4) TSR 3.3.4.3(3)	0 - 1.0 g
b. 0-XT-52-75B Reactor Bldg. El. 757	1	TSR 3.3.4.1(2) TSR 3.3.4.2(4) TSR 3.3.4.3(3)	0 - 1.0 g
c. 0-XT-52-75D D/G Bldg. El. 742	1	TSR 3.3.4.1(2) TSR 3.3.4.2(4) TSR 3.3.4.3(3)	0 - 1.0 g
2. Triaxial Strong Motion Accelerograph			
a. 0-XR-52-80 Aux. Cont. Room. El 757	1	TSR 3.3.4.1(2) TSR 3.3.4.2(4) TSR 3.3.4.3(3)	0 - 2.0 g

(1) With associated acceleration triggers, and control room indication on 0-XR-52-82A, -82B, -83.
(2) Except acceleration trigger.
(3) Includes acceleration trigger.
(4) Except setpoint verification.
(5) Includes recording and analyzing components on 0-R-113.

TR 3.3 INSTRUMENTATION

TR 3.3.5 Turbine Overspeed Protection

TR 3.3.5 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

-----NOTES-----

1. Not applicable to MODES 2 and 3 when all main steam isolation valves are closed and all other steam flow paths to the turbine are isolated.
2. TR 3.0.4 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One high pressure turbine steam inlet valve inoperable.	A.1.1 Verify the two high pressure turbine steam inlet valves on the same steam chest which are opposite the inoperable valve are OPERABLE.	6 hours
	AND	
	A.1.2 Restore inoperable valve to OPERABLE status.	72 hours
	OR	
	A.2.1 Verify the two high pressure turbine steam inlet valves on the same steam chest which are opposite the inoperable valve are OPERABLE.	6 hours
	AND	
	A.2.2 Remove the turbine from service by closing all the high pressure turbine steam inlet valves.	78 hours
	OR	
	A.3 Close MSIVs.	78 hours

(continued

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.6.1	Perform CHANNEL CHECK.	24 hours
TSR 3.3.6.2	Perform CHANNEL OPERATIONAL TEST.	31 days
TSR 3.3.6.3	Perform CHANNEL CALIBRATION.	18 months

TR 3.3 INSTRUMENTATION

TR 3.3.7 Plant Calorimetric Measurement

TR 3.3.7 The Leading Edge Flow Meter (LEFM) shall be used for the completion of SR 3.3.1.2.

APPLICABILITY: MODE 1 > 15% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LEFM not available.	A.1 Restore LEFM to available.	Prior to next performance of SR 3.3.1.2
B. Required Action and associated Completion Time not met.	B.1 Ensure THERMAL POWER ≤ 98.6% RTP (3411 MWt).	Prior to performance of SR 3.3.1.2
	<u>AND</u>	
	B.2 Perform SR 3.3.1.2 based on feedwater venturis calorimetric.	As required by SR 3.3.1.2
	<u>AND</u>	
	B.3 Maintain THERMAL POWER ≤ 98.6% RTP (3411 MWt)	Until LEFM is restored to available status and SR 3.3.1.2 is performed using LEFM.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.7.1	Verify availability of the LEFM using the self-diagnostics feature indicated by the LEFM Normal/Alert/Fail status indication, as displayed on the plant computer system, is not in Fail status.	Prior to performance of SR 3.3.1.2.

5.0 ADMINISTRATIVE CONTROLS

5.1 Technical Requirements (TR) Control Program

This Program provides a means for controlling changes or additions to the Technical Requirements and their Bases.

5.1.1 Changes or additions to the Technical Requirements Manual (TRM) shall be made under appropriate administrative controls and reviews.

5.1.2 Licensees may make changes or additions to the TRM without prior NRC approval provided the changes have been determined not to be candidates for inclusion in the Technical Specifications (TS) and not to require NRC approval pursuant to 10 CFR 50.59. The changes to the TRM shall include the following:

a. Include screening the change against the criteria contained in 10 CFR 50.36(c)(2)(ii).

b. The change does not require NRC approval pursuant to 10 CFR 50.59.

5.1.3 The Technical Requirements Control Program shall contain provisions to ensure that changes or additions to the TRM are accurately reflected in the FSAR as appropriate.

5.1.4 Proposed changes or additions that do not meet the criteria of 5.1.2 shall be reviewed and approved by the NRC prior to implementation. Changes or additions to the TRM implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

BASES

TR (continued)

b. A flow path from an OPERABLE RWST through a charging pump to the RCS.

APPLICABILITY

The OPERABILITY of one boron injection flow path ensures that this system is available for reactivity control while in MODES 4, 5, and 6.

Boron injection flow paths for MODES 1, 2, and 3 are covered in Technical Requirement 3.1.2, "Boration Systems Flow Paths, Operating".

ACTIONS

A.1 and A.2

With the Boration Systems flow path OPERABILITY requirements not met, or the Boration Systems flow path not capable of being powered by an OPERABLE emergency power source, the plant must be placed in a condition where negative reactivity addition is not required. This is accomplished by suspending all CORE ALTERATIONS and positive reactivity additions immediately. One boron injection flow path is required to meet the TR and to ensure that negative reactivity control is available during MODES 4, 5, and 6. Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition.

The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.1.1

This surveillance verifies the temperature of the areas containing portions of the flow path from the boric acid tanks is $\geq$ 63°F (value does not account for instrument error). This ensures that the high concentration of boric acid in the storage tanks is not allowed to precipitate due to cooling.

The Surveillance is modified by a note stating that the surveillance is required only if a flow path from the boric

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.1.1 (continued)

acid storage tanks is required OPERABLE. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

TSR 3.1.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in the Boration System flow path provides assurance that the proper flow paths exist for Boration System operation. This TSR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This TSR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This TSR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

REFERENCES

1. Watts Bar FSAR, Section 9.3.4, "Chemical and Volume Control System."

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

3. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar Nuclear Plant, Unit 1, Revision 00, April 1993."

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.2.1

This surveillance verifies the temperature of the required flow path from the boric acid tanks to be at least 63°F (value does not account for instrument error). This ensures that the high concentration of boric acid in the storage tanks is not allowed to precipitate due to cooling.

The surveillance is modified by a note stating that the surveillance is required only if the flow path from the boric acid storage tanks is used as one of the two required flow paths. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

TSR 3.1.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the Boration System flow path provides assurance that the proper flow paths exist for Boration System operation. This TSR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This TSR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This TSR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

TSR 3.1.2.3

This surveillance demonstrates that each automatic valve in the flow path actuates to its required position on an actual or simulated actuation signal. The 18-month Frequency was developed considering it is prudent that this surveillance only be performed during a plant outage. This is due to the plant conditions needed to perform the TSR and the potential for unplanned plant transients if the TSR is performed with the reactor at power.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.2.4

Verification that the flow path from the boric acid tanks delivers at least 35 gpm (value does not account for instrument error) to the RCS demonstrates that gross degradation of the boric acid transfer pumps, crystallization of boric acid in the system, and other hydraulic component problems have not occurred.

The 18-month Frequency was developed considering it is prudent that this surveillance only be performed during a plant outage. This is due to the plant conditions needed to perform the TSR and the potential for unplanned plant transients if the TSR is performed with the reactor at power.

REFERENCES

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

2. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis For Watts Bar Nuclear Plant, Unit 1," Revision 00, April 1993.

BASES (continued)

APPLICABILITY

The OPERABILITY of one borated water source in the required boron injection flow path ensures that this system is available for reactivity control while in MODES 4, 5, and 6.

Borated water source OPERABILITY requirements for MODES 1, 2, and 3 are covered in Technical Requirement 3.1.6, "Borated Water Sources, Operating."

ACTIONS

A.1 and A.2

If the required borated water source is inoperable, the plant must be placed in a condition where negative reactivity addition is not required. This is accomplished by suspending all CORE ALTERATIONS and positive reactivity additions immediately. One borated water source is required to meet the TR and to ensure that negative reactivity control is available during MODES 4, 5, and 6. Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition.

The immediate Completion Time is consistent with the required times for actions requiring prompt attention.

TECHNICAL SURVEILLANCE REQUIREMENTS

The Notes in the Technical Surveillance Requirements state that TSR 3.1.5.1, TSR 3.1.5.2, and TSR 3.1.5.3 are only required to be performed if the RWST is the required borated water source, and TSR 3.1.5.4, TSR 3.1.5.5, and TSR 3.1.5.6 are only required to be performed if the BASS is the required borated water source.

TSR 3.1.5.1

This surveillance requires verification every 24 hours that the RWST temperature is greater than or equal to 60°F (value does not account for instrument error). The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 60°F temperature limit and has been shown to be acceptable through operating experience. The TSR is modified by a Note which eliminates the requirement to perform this surveillance when ambient air temperature

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.5.1 (continued)

is greater than or equal to 60°F. With ambient air temperature greater than 60°F, the RWST solution temperature should not decrease below this limit, therefore, monitoring is not required.

TSR 3.1.5.2

This surveillance requires verification every 7 days that the boron concentration of the RWST is ≥ 2,500 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 8.0 and 10.5. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.5.3

This surveillance requires verification every 7 days that the RWST borated water volume is ≥ 62,900 gallons (value does not account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between 8.0 and 10.5. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the RWST volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience. The 62,900 gallon volume requirement includes 11,100 gallons for shutdown margin, adjustments for minimum safety limit level in the RWST, and adjustments for instrument error.

TSR 3.1.5.4

This surveillance requires verification every 24 hours that the Boric Acid Tank (BAT) solution temperature is ≥ 63°F (value does not account for instrument error). This ensures that the concentration of boric acid in the BAT is not allowed to precipitate due to cooling. The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 63°F temperature limit.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.1.5.5

This surveillance requires verification every 7 days that the boron concentration of the BAT is between 6,120 ppm and 6,990 ppm. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 8.0 and 10.5. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.5.6

This surveillance requires verification every 7 days that the BAT borated water volume is ≥ 3,800 gallons (value does not account for instrument error). This borated water volume is sufficient to provide an adequate SDM and also ensure a pH value between 8.0 and 10.5 This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

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

2. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar, Unit 1," Revision 00, April 1993.

3. TVA Calculation, EPM-PDM-071197, Revision 0, "Boric Acid Concentration Analysis for BAT and RWST."

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.6 Borated Water Sources, Operating

BASES

BACKGROUND	A description of the Boration System Flow Paths, which include borated water sources is provided in the Bases for Technical Requirement 3.1.1, "Boration System Flow Paths, Shutdown."
APPLICABLE SAFETY ANALYSES	The boration subsystem is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of a malfunction of the Chemical and Volume Control System, which causes a boron dilution event, the automatic response, or that required by the operator, is to close the appropriate valves in the reactor makeup system. This action is required before the SDM is lost. Operation of the boration subsystem is not assumed to mitigate this event (Ref. 1). OPERABILITY of the charging pumps, the RWST, and the appropriate flow paths is required as part of the Emergency Core Cooling System (ECCS). The Technical Specifications for the ECCS address the requirements of these components.
TR	TR 3.1.6 requires a Boric Acid Storage System (BASS) and the Refueling Water Storage Tank (RWST) to be OPERABLE as required by TR 3.1.2. This is a requirement during MODES 1, 2, and 3 to accomplish (1) normal makeup, (2) chemical shim reactivity control, and (3) miscellaneous fill and transfer operations.
APPLICABILITY	The OPERABILITY of borated water sources (as required by TR 3.1.2) in the required boron injection flow paths ensures that this system is available for reactivity control while in MODES 1, 2, and 3. Borated water source OPERABILITY requirements for MODES 4, 5 and 6 are covered in Technical Requirement 3.1.5, "Borated Water Sources, Shutdown."

(continued)

BASES (continued)

ACTIONS

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

With the BASS inoperable, action must be taken to restore the BASS to OPERABLE status within 72 hours. The Completion Time of 72 hours to perform Required Action A.1 is reasonable based upon the typical time necessary to effect repairs and the redundant capabilities afforded by the OPERABLE borated water source.

If the BASS cannot be restored to OPERABLE status the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 and by borating to a SDM equivalent to at least 1% Δk/k at 200°F in 6 additional hours (78 hours total time). It is also required that the BASS be restored to OPERABLE status in an additional 7 days (246 hours total time).

The 6 additional hours to perform Required Actions A.2.1 and A.2.2 are reasonable and based on operating experience to reach MODE 3 and the required SDM from full power operation in an orderly manner and without challenging plant systems. The 7 day Completion Time per Required Action A.2.3 is based on the low probability of an event occurring during this time period, and the consideration that the remaining borated water sources can provide the required capability.

B.1

If the Required Actions and associated Completion Times of Condition A are not met, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in MODE 4 within 6 hours. The allowed Completion Time is reasonable and based on operating experience to reach required plant conditions in an orderly manner and without challenging plant systems.

C.1

With the RWST boron concentration or borated water temperature not within limits, action must be taken within 8 hours to restore the RWST to OPERABLE status. This 8-hour limit was developed considering the time required to change either the boron concentration or water temperature. The Completion Time is consistent with Technical Specification 3.5.4, "Refueling Water Storage Tank."

(continued)

BASES

ACTIONS (continued)

D.1

With the RWST inoperable for reasons other than Condition C (e.g. water volume), it must be restored to OPERABLE status within 1 hour. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting two of the boration system flow paths. The Completion Time is consistent with Technical Specification 3.5.4, "Refueling Water Storage Tank."

E.1 and E.2

If the Required Actions and associated Completion Times of Condition C or D are not met, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Time is reasonable and based on operating experience to reach required plant conditions in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.6.1

The limits assumed in the accident analysis band for the RWST borated water temperature are ≥ 60°F and ≤ 105°F (values do not account for instrument error). This surveillance requires verification of the water temperature limits every 24 hours. This is frequent enough to identify a temperature change that would approach either temperature limit and has been shown to be acceptable through operating experience.

The TSR is modified by a Note which eliminates the requirement to perform this surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST solution temperature should not exceed the limits.

TSR 3.1.6.2

This surveillance requires verification every 7 days that the boron concentration of the RWST is within the required band. This ensures the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.6.2 (continued)

boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7-day Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.6.3

This surveillance requires verification every 7 days that the RWST borated water volume is within the required limit of ≥ 370,000 gallons (value does not account for instrument error). This will ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

TSR 3.1.6.4

This surveillance requires verification every 24 hours that the Boric Acid Tank (BAT) solution temperature is ≥ 63°F (value does not account for instrument error). This ensures that the concentration of boric acid in the BAT is not allowed to precipitate due to cooling. The Frequency of 24 hours for performance of the surveillance is frequent enough to identify a temperature change that would approach the 63°F temperature limit and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

TSR 3.1.6.5

This surveillance requires verification every 7 days that the boron concentration of the BAT is in accordance with Figure 3.1.6 of TR 3.1.6. This boron concentration is sufficient to provide an adequate SDM and also ensure a pH value between 8.0 and 10.5. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. Since the BAT volume is normally stable, a

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.6.5 (continued)

7-day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

TSR 3.1.6.6

This surveillance requires verification every 7 days that the BAT borated water volume is in accordance with Figure 3.1.6 (the values listed on the figure do not account for instrument error). This borated water volume at the boron concentration specified in TSR 3.1.6.5 is sufficient to provide an adequate SDM. Since the BAT volume is normally stable, a 7-day Frequency to verify borated water volume is appropriate and has been shown to be acceptable through operating experience.

This surveillance has been modified by a NOTE stating that the surveillance is only required if the BAT is used as one of the required borated water sources for TR 3.1.2.

The maximum expected boration capability requirement occurs near EOL from full power peak xenon conditions and requires borated water from a boric acid tank in accordance with Figure 3.1.6, and additional makeup from either (1) the common boric acid tank and/or batching tank, or (2) a maximum of 23,000 gallons of 2,500 ppm borated water from the refueling water storage tank. With the refueling water storage tank as the only borated water source, a maximum of 61,000 gallons of 2,500 ppm borated water is required.

REFERENCES

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

2. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar Nuclear Plant, Unit 1," Revision 00, April 1993.

3. TVA Calculation, EPM-PDM-071197, Revision 0, "Boric Acid Concentration Analysis For BAT and RWST."

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.1.1

TSR 3.3.1.1 demonstrates that the RTS RESPONSE TIME of each reactor trip function is within the limits listed in Table 3.3.1-1 of the TR. This ensures that the time delays assumed in the safety analyses are not exceeded. Each train's response must be verified every 18 months on a STAGGERED TEST BASIS (i.e., Train A at 18 months after initial startup, Train B at 36 months, and then Train A again). Response times cannot be determined during plant operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed on the 18-month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Table 3.3.1-1 of this TR specifies the RESPONSE TIMES for the RTS.

REFERENCES

1. Watts Bar FSAR, Section 15.1.3. "Trip Points and Time Delays To Trip Assumed in Accident Analyses."
2. Watts Bar FSAR, Section 7.0 "Instrumentation and Controls."
3. Watts Bar FSAR, Section 15.0 "Accident Analyses."
4. Watts Bar Technical Specifications (Unit 1), Section 3.3.1, "Reactor Trip Instrumentation" and Bases for 3.3.1.

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Features Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based upon 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. A detailed Background for ESFAS is given in Reference 1. This TR covers only RESPONSE TIME testing.

The ESFAS RESPONSE TIME is defined as the interval required for the ESF sequence to be initiated subsequent to the time that the appropriate variables exceed the setpoints. This definition is augmented in the Standard Technical Specifications to include automatic system lineups and diesel generator starting and sequence loading delays. The ESF sequence is initiated by the output of the ESFAS, which is by the operation of the dry contacts of the slave relays (600 series relays) in the output cabinets of the Solid State Protection System (SSPS). The RESPONSE TIMES listed in Table 3.3.2-1 of this TR include the interval of time which will elapse between the time the parameter as sensed by the sensor exceeds the safety setpoint and the time the SSPS slave relay dry contacts are operated. The values listed are maximum allowable values consistent with the safety analyses and this Technical Requirement and are systematically verified during plant preoperational startup tests. For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of sequential tests such that the entire response time is measured. The overall ESFAS RESPONSE TIMES are listed in this TR.

The ESFAS is always capable of having response time tests performed using the same methods as those tests performed during the preoperational test program or following significant component changes (Ref. 2).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The required channels of ESFAS Instrumentation provide plant protection in the event of any of the analyzed accidents. The accident analyses described in Reference 3 take credit for operation of ESF systems during DBAs. The safety analyses applicable to each ESFAS function are discussed in the bases for the Technical Specifications, B 3.3.2 (Ref. 1), B 3.3.5 (Ref. 4) and B 3.3.6 (Ref. 5).
TR	OPERABILITY requirements for ESFAS Instrumentation are specified in Technical Specifications, LCO 3.3.2, 3.3.5 and 3.3.6. TR 3.3.2 requires the ESFAS Instrumentation of Table 3.3.2-1 of the TR to be OPERABLE with RESPONSE TIMES as shown in the table. RESPONSE TIMES must be within the specified limits for the affected instruments to be considered OPERABLE.
APPLICABILITY	Applicable MODES for the specific ESFAS Instrumentation are delineated in Table 3.3.2-1 of Reference 1; in the Applicability of Reference 4; and in Table 3.3.6-1 of Reference 5. The bases for Applicability of each function is included in References 1, 4 and 5.
ACTIONS	<u>A.1</u> The required Actions for inoperable instruments is found in Reference 1. With one or more RESPONSE TIMES outside the specified limits, the affected instrument(s) must be considered inoperable and the appropriate Action referenced in Table 3.3.2-1 of Reference 1; the Actions of Reference 4; or the appropriate Action of Table 3.3.6-1, must be taken. The bases for these actions is found in References 1, 4 and 5.
TECHNICAL SURVEILLANCE REQUIREMENTS	<u>TSR 3.3.2.1</u> TSR 3.3.2.1 demonstrates that the ESFAS RESPONSE TIME of each ESFAS function is within the limits listed in Table 3.3.2-1 of the TR. This ensures that the time delays assumed in the safety analyses are not exceeded. Response

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.2.1 (continued)

time tests are conducted on an 18-month STAGGERED TEST BASIS. The 18-month Frequency was developed considering it was prudent that these Surveillances only be performed during a plant outage. This was due to the plant conditions needed to perform the Surveillance and the potential for unplanned plant transients if the Surveillance is performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18-month Frequency.

Table 3.3.2-1 of this TR specifies the RESPONSE TIMES for the ESFAS Instrumentation.

REFERENCES

1. Watts Bar Technical Specifications (Unit 1), Section 3.3.2, "Engineered Safety Features Actuation System Instrumentation" and Bases for 3.3.2.

2. Watts Bar FSAR, Section 7.3.1.2.6, "Minimum Performance Requirements."

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

4. Watts Bar Technical Specifications (Unit 1), Section 3.3.5, "LOP Diesel Generator Start Instrumentation" and Bases for 3.3.5.

5. Watts Bar Technical Specifications (Unit 1), Section 3.3.6, "Containment Vent Isolation Instrumentation" and Bases for 3.3.6.

BASES (continued)

ACTIONS

A.1

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

Inoperable Movable Incore Detection Systems cannot be used for recalibration of the Excore Neutron Flux Detection System, or monitoring the QPTR or measurement of $F_{\Delta H}^{N}$, $F_Q(Z)$ and F_{XY}. Therefore, the Required action A.1 prohibits the use of the inoperable system for the above applicable monitoring or calibration functions.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.3.1

The Movable Incore Detector System must be demonstrated OPERABLE at least once per 24 hours by the setting of each detector's operating voltage. The operating voltage is set by determining the operating region for each detector after inserting it into a high flux region of the core. The acceptability of each detector is verified by the performance of a detector drift check. The operating voltage must be determined prior to using the Movable Incore Detector System for recalibration of the Excore Neutron Flux Detection System, or monitoring the QPTR, or measurement of $F_{\Delta H}^{N}$, $F_Q(Z)$ and F_{XY}. This surveillance ensures that the measurements obtained from use of this system accurately represents the spatial neutron flux distribution of the core. The Frequency of 24 hours has been established, based on engineering judgment, and has been shown to be acceptable through operating experience.

REFERENCES

1. WCAP-8648, "Excore Detector Recalibration Using Quarter-core Flux Maps," June 1976.

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

B 3.3 INSTRUMENTATION

B 3.3.4 Seismic Instrumentation

BASES

BACKGROUND

The seismic instrumentation is made up of several instruments such as accelerometers, an accelerograph, recorders, etc. These instruments are placed in several appropriate locations throughout the plant in order to provide data on the seismic input to containment, data on the frequency, amplitude and phase relationship of the seismic response of the containment structure, and data on the seismic input to other Seismic Category I structures (Ref. 1).

The seismic instrumentation is used to promptly determine the nature and severity of a seismic event and to predict the impact (i.e., potential for damage) on nuclear power plant features which are important to safety. This is required to permit comparison of the measured response to that used in the design basis for the unit to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Reference 1.

The original seismic instrumentation was replaced with state of the art digital instrumentation in order to permit application of EPRI OBE exceedance criteria delineated in References 4 and 5. Use of these criteria is permitted by Reference 6 provided that upgraded instrumentation is used. The replacement instrumentation is capable of recording a seismic event and performing appropriate analyses of the recorded data to provide an immediate basis for determining whether an OBE exceedance has occurred. Reference 6 directs that this information must be evaluated within 4 hours after an event and a walkdown of critical plant features must be accomplished within 8 hours after an event in order to make a determination as to whether a plant shutdown in warranted.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and to determine the impact on those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit to determine if plant equipment inspection is required pursuant to Appendix A of 10 CFR part 100 prior to restart. Seismic risks which appear as dominant sequences in PRAs occur for very severe earthquakes with magnitudes which are a factor of two or three above the Safe Shutdown Earthquake and Design Basis Earthquake. The Seismic Instrumentation System was not designed to function or to provide comparative information for such severe earthquakes. This instrumentation is more pertinent to determining the need to shut down following a seismic event and the ability to restart the plant after seismic events which are not risk contributors, and is therefore not of prime importance in risk dominant sequences (Ref. 2).

(continued)

BASES (continued)

TR	TR 3.3.4 requires that the seismic monitoring instrumentation which is shown in Table 3.3.4-1 shall be OPERABLE. This requirement ensures that an assessment can be made of the effects on the plant of earthquakes which may occur that exceed the design basis spectra for the Operating Basis Earthquake (Ref. 3).
APPLICABILITY	Since the possibility of earthquakes is not MODE dependent. OPERABILITY of the seismic instrumentation is required at all times. The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

The determination as to whether an OBE exceedance has occurred is made by comparing the calculated spectra for the event with the applicable design basis spectra for that building and location. Reference 6 requires that this determination be made considering the data from instruments located on the Containment foundation. Therefore, the exceedance determination for WBN will be made using event data from 0-XT-52-75A in the Containment annulus. Data from this instrument is recorded at panel 0-R-113, which also contains the computer used to calculate the spectral content and the alarm panel used to annunciate in the control room. These devices are the key components used to detect the event and make a shutdown determination. With one or more of these required seismic monitoring instruments inoperable for more than 30 days, a report must be submitted to the Commission in accordance with 10 CFR 50.4. The report is to outline the cause of the malfunction and the plans for restoring the inoperable instruments to OPERABLE status.

With one or more of the remaining seismic instruments inoperable for more than 60 days, a report must also be submitted as noted above. A longer period of inoperability is allowed for these instruments since they are used only for evaluating plant condition following an event and not for input to the shutdown decision.

The Completion Time of 10 days to perform Required Action A.1 is reasonable and based upon the typical time necessary to prepare and submit a report to the NRC.

B.1 and B.2

When one or more seismic monitoring instruments actuate during a seismic event with greater than or equal to 0.01g ground acceleration, all of the Required Actions under Condition B must be completed. The data retrieved from the actuated instruments must be analyzed to determine the magnitude of the vibratory ground motion. The replacement digital instrumentation provides the capability to analyze the event data onsite and generate event spectra to be used in determining whether an OBE exceedance

continued

BASES

ACTIONS (continued)

has occurred. If an OBE exceedance has occurred, Reference 6 directs that this evaluation should occur within 4 hours after the event. Reference 6 also requires performance of a limited scope walkdown per Reference 7 to determine the extent of actual damage within 8 hours following the event. The information provided by this walkdown and the spectral analysis are to be used in making a determination as to whether to proceed with plant shutdown.

B.3 and B.4

Each actuated monitoring instrument must be restored to OPERABLE status within 24 hours. Within 10 days of the actuation, a CHANNEL CALIBRATION must be performed on each actuated monitoring instrument. The Completion Time of 10 days to perform Required Action B.4 is reasonable and is based on engineering judgment.

B.5 and B.6

Subsequent analysis must then be performed using data from the remaining seismic monitoring instruments to evaluate the plant response in comparison with previously generated design basis spectra at the locations of those instruments. A report must be sent to the NRC in accordance with 10CFR 50.4. This report is to describe the magnitude, frequency spectrum, and resultant effect upon unit features important to safety. The Completion Time of 14 days to perform Required Action B.5 and B.6 is reasonable and based upon the typical time necessary to analyze data and prepare a report.

TECHNICAL SURVEILLANCE REQUIREMENTS

The SRs for each seismic monitoring Function are identified by the SRs column of Table 3.3.4-1.

A Note has been added to the TSRs to clarify that Table 3.3.4-1 determines which SRs apply to which seismic monitoring instruments.

Performance of a CHANNEL CHECK on the seismic instrumentation once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a check of external system status indications that the seismic monitoring equipment is in a state of readiness to properly function should an earthquake occur. A CHANNEL CHECK will detect gross system failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL OPERATIONAL TEST.

The Surveillance Frequency of 31 days is based on operating experience related to to instrumentation systems, which demonstrates that gross instrumentation system failure in any 31-day interval is a rare event. The CHANNEL CHECK supplements the loss of power annunciation for the equipment in the auxiliary instrument room. The equipment in the auxiliary control room doe not have a loss of power alarm but only provides supplemental data.

(continued

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.3.4.2

A CHANNEL OPERATIONAL TEST is to be performed on each required channel to ensure the entire channel will perform the intended function. A CHANNEL OPERATIONAL TEST is the comparison of the response of the instrumentation, including all components of the instrument, to a known signal. Although the seismic trigger is functionally checked, its setpoint is not verified. The Surveillance Frequency of 184 days is based upon the known reliability of the monitoring instrumentation and has been shown to be acceptable through operating experience.

TSR 3.3.4.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor by comparing the response of the instrument to a known input on the sensor. This test verifies the capability of the seismic instrumentation to correctly determine the magnitude of a seismic event and evaluate the response of those features important to safety. The Surveillance Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. Regulatory Guide 1.12, "Instrumentation for Earthquakes," Revision 1, April 1974.
2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.
3. Watts Bar FSAR, Section 3.7.4, "Seismic Program."
4. EPRI NO-5930, July 1988, "A Criterion for Determining Exceedance of the Operating Basis Earthquake."
5. EPRI TR-104239, June 1994, "Seismic Instrumentation in Nuclear Power Plants for Response to OBE Exceedance: Guideline for Implementation."
6. Regulatory Guide 1.166, "Pre-Earthquate Planning And Immediate Nuclear Power Plant Operator Postearthquake Actions," Revision 0, March 1997.
7. EPRI NP-6695, December 1989, "Guidelines for Nuclear Plant Response to an Earthquake."

B 3.3 INSTRUMENTATION

B 3.3.5 Turbine Overspeed Protection

BASES

BACKGROUND

Three types of overspeed protection mechanisms are provided to isolate main steam to the turbo-generator when the rated operating speed of 1800 rpm is exceeded. During normal speed-load control, the Analog Electro Hydraulic (AEH) Overspeed Protection Control (OPC) which is set at 1854 rpm (103 percent of rated speed) will rapidly close the governor and interceptor valves in case of an overspeed condition. Rotational speed is then maintained below this runback setpoint by moving the interceptor valves between the closed and open position until the reheater steam (steam between the high pressure turbine exhaust and the low pressure turbines) is dissipated. If the AEH control system is in the automatic mode, the governor valves will take over speed control and will maintain reference speed. However, if the AEH control system is in the manual mode (normally only at low power levels during startup), the turbine generator will coast down to turning gear operation, if no operator action is taken.

If for some reason the AEH OPC control system does not function and the turbine speed increases to 1980 rpm (110 percent of rated speed), the mechanical overspeed mechanism will trip close all steam valves (throttle, governor, reheat, stop, and interceptor valves and prevent the turbine speed from exceeding 120 percent of rated speed. The unit will then coast down to turning gear operation.

In addition to these two control systems, an independent electrical overspeed trip is provided in the Analog Electro Hydraulic (AEH) Control System. If the turbine generator speed increases to 1998 rpm (111 percent of rated speed), all steam valves (as listed in the previous paragraph) will be tripped closed. This trip will be actuated by a contact output from the AEH controller which energizes a trip solenoid in the autostop oil fluid lines. Again, during the overspeed condition, turbine speed will remain below 120 percent of rated speed. The unit will then coast down to turning gear operation. (Ref. 1)

(continued)

B 3.3 INSTRUMENTATION

B 3.3.7 Plant Calorimetric Measurement

BASES

BACKGROUND

The predominant contribution to the secondary plant calorimetric measurement uncertainty is the uncertainty associated with the feedwater flow measurement. Traditionally, a differential pressure (ΔP) transmitter across a venturi in each main feedwater line has been used to provide the feedwater flow. However, the venturis are subject to fouling and the uncertainty associated with the flow derived from the ΔP indication can be large and increases as the flow deviates from the "optimum" conditions for which the ΔP transmitter was calibrated.

More recently, Leading Edge Flow Meters (LEFMs) have been used to provide the feedwater flow input to the secondary plant calorimetric measurement. The uncertainty associated with the LEFM is relatively small and is independent of the actual feedwater flow.

Most of the original safety analyses supporting plant operation (including LOCA analyses required by 10CFR50 Appendix K) were performed at a maximum power level of 3411 MWt plus an allowance for the secondary plant calorimetric uncertainty assumed to be 2% above rated thermal power. Hence, it is possible to support operation at a higher power level while remaining within the original analyses.

The RATED THERMAL POWER for Unit 1 is 3459 MWt which represents an increase of 1.4% RTP from the originally licensed value of 3411 MWt. This uprate is based on reduced uncertainties associated with the secondary plant calorimetric measurement that is attained through the use of the LEFM Check supplied by Caldon, Inc. Many of the accident analyses are performed at 102% of 3411 MWt, or 3479 MWt, where the 2% RTP is an allowance for the uncertainty associated with the power calorimetric measurement. With the LEFM Check, the power calorimetric measurement uncertainty is less than 0.6% RTP. Without performing new Appendix K accident analyses, the LEFM Check can be used to allow the plant to be operated at a redefined 100% RTP of 3459 MWt.

(continued)

BASES

BACKGROUND (continued)

However, this allowance is predicated on the availability of the LEFM Check for performance of the calorimetric measurement. When the LEFM Check is unavailable, the uncertainties associated with the feedwater venturi-based measurement (2% RTP) must be used to ensure compliance with the safety analysis value of the core power of 3479 MWt.

Surveillance Requirement SR 3.3.1.2 requires the performance of a comparison of the results of the calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output. SR 3.3.1.2 note 1 requires that the NIS channels be adjusted if the absolute difference is > 2% RTP.

SR 3.3.1.2 is required to be performed every 24 hours (daily). At that time, the NIS indication must be normalized to indicate within at least ± 2% RTP of the calorimetric measurement. The plant may then be run for the next 24 hour period using this normalized NIS indication, such that the calorimetric power does not exceed 100% RTP. Although the calorimetric power indication may be monitored continuously for control of the unit power, the calorimetric power indication is not required to be consulted again until the daily calorimetric comparisons of the NIS indication are performed.

The following general guidance is provided for operation of WBN Unit 1:

1) When the LEFM Check is available, the plant should be operated in a manner consistent with the LEFM Check based calorimetric measurement and at 3459 MWt (100% RTP).

2) If the LEFM Check is unavailable, the plant may be operated at 3459 MWt (100% RTP) using the NIS indication until the next performance of SR 3.3.1.2 is due.

3) If the LEFM Check based calorimetric measurement is unavailable at the time SR 3.3.1.2 is due, the normalized feedwater venturi-based calorimetric measurement should be used for the performance of SR 3.3.1.2. However, to maintain consistency with the uncertainty analysis, the maximum allowable power should be reduced to 3411 MWt or 98.6% RTP. Either the NIS indication or the normalized feedwater venturi-based calorimetric power indication may be used to control the unit power.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	Each of the analyzed accidents are evaluated for the range of power levels over which the reactor is allowed to be operated. Typically, the analyses are most limiting when initiated from a higher power level. For Unit 1, the majority of the original analyses were performed for a core power of at least 3411 MWt, plus an allowance for the secondary power calorimetric measurement of 2% RTP. In general, these same analyses are used to support the revised RATED THERMAL POWER definition of a core power of 3459 MWt. With the application of a 0.6% RTP uncertainty (based on the use of the LEFM Check feedwater flow input into the secondary calorimetric measurement), the analyses are evaluated at a power level of 3479 MWt. Analyses that use statistical methods, such as the analysis of the dropped RCCA event, are explicitly evaluated for operation at 3411 MWt with a 2% RTP uncertainty allowance and for operation at 3459 MWt with a 0.6% RTP uncertainty allowance. The setpoints for those functions of the Reactor Protection System that are based on percentage of power (i.e., the NIS) have been calculated based on analytical margins available at the 3459 MWt definition of 100% RTP. Operation back at 3411 MWt does not require these setpoints to be adjusted.
TR	The TR requires the LEFM Check to be used for the completion of the daily secondary plant calorimetric measurement required in SR 3.3.1.2. The use of the LEFM Check ensures that the basis for operation at the RATED THERMAL POWER of 3459 MWt is maintained.
APPLICABILITY	The requirement to use the LEFM Check for the performance of the secondary plant calorimetric measurement required by SR 3.3.1.2 is applicable to Unit 1 in mode 1 above 15% RTP, consistent with the applicability of SR 3.3.1.2.

(continued)

BASES (continued)

ACTIONS

A.1

If the LEFM becomes unavailable during the intervals between performance of SR 3.3.1.2, plant operation may continue using the power indications from the NIS system. However, in order to remain in compliance with the bases for operation at a RATED THERMAL POWER of 3459 MWt, the LEFM must be returned to service prior to performance of SR 3.3.1.2

B.1, B.2, and B.3

If the Required Action and associated Completion Time of Condition A is not met (i.e., the LEFM has not been returned to service prior to the performance of SR 3.3.1.2), Condition B is entered. Required Action B.1 requires that the reactor power be reduced to, or maintained at, a power level less than or equal to 98.6% RTP (3411 MWt). This power reduction is performed prior to performing SR 3.3.1.2, in order to remain within the plant's design bases immediately upon performance of SR 3.3.1.2.

Required Action B.2 directs the performance of SR 3.3.1.2 using the feedwater venturi indications of feedwater flow. Once SR 3.3.1.2 is performed using the feedwater venturi indications of feedwater flow, the required power uncertainty is 2% RTP. In order to maintain compliance with the safety analyses, it is necessary to operate the plant at a maximum core thermal power of 3411 MWt.

Required Action B.3 serves as a reminder that the core power is to be maintained at a value less than or equal to 3411 MWt until the LEFM is returned to service and SR 3.3.1.2 has been performed using the LEFM indication of feedwater flow. Once SR 3.3.1.2 has been performed using the LEFM, the plant can again be operated at 3459 MWt.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.7.1

TSR 3.3.7.1 requires that the availability of the LEFM be verified prior to its use for the performance of SR 3.3.1.2. The self diagnostic features of the LEFM Check are used for this surveillance. If the LEFM Normal/Alert/Fail status indication as displayed on the plant computer system is not in Fail status, it is considered operable.

REFERENCES

1. License Amendment Request TVA-WBN-TS-00-06, increases the licensed power for operation of WBN Unit 1 to 3459 MWt, Docket No. 50-390.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Ice Bed Temperature Monitoring System

BASES

BACKGROUND	The Ice Bed Temperature Monitoring System consists of Resistance Temperature Detectors (RTDs) which are located in various parts of the ice condenser. They serve to verify the attainment of a uniform equilibrium temperature in the ice bed and to detect general gradual temperature rise in the cooling system if breakdown occurs. Forty-seven RTDs are mounted on ice bed probes which are located throughout the ice bed. These 47 RTDs tie into a temperature scanner unit, located in the Incore Instrument Room. The scanner multiplexes the ice condenser RTD's signals to a Westronics recorder in the Main Control Room. There are also six temperature switches located at various points in the ice bed to serve as backup indication should the scanner unit or recorder fail to operate. These inputs provide an alarm on the control room annunciator panel should the ice bed temperature exceed preset value (Ref. 1). In addition, the 47 RTDs can be read from the local ice condenser temperature monitoring panel.
APPLICABLE SAFETY ANALYSES	The ice condenser is a passive device requiring only maintenance of the ice inventory in the ice bed. As such there are no actuation circuits or equipment which are required for the ice condenser to operate in the event of a Loss of Coolant Accident (LOCA). The Ice Bed Temperature Monitoring System serves only to monitor the ice bed temperature. Since the ice bed has a very large thermal capacity, postulated off-normal conditions can be successfully tolerated for a week to two weeks. Therefore, the Ice Bed Temperature Monitoring System provides an early warning of any incipient ice condenser temperature anomalies. The Ice Bed Temperature Monitoring System is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. Based on the PRA Summary Report (Ref. 2), the Ice Bed Temperature Monitoring System has not been identified as a significant risk contributor.

(continued)

BASES (continued)

TR

TR 3.6.1 states that the Ice Bed Temperature Monitoring System shall be OPERABLE with at least two OPERABLE RTD channels in the ice bed at each of three basic elevations: 10'6", 30'9", and 55' above the floor of the ice condenser, for each one-third of the ice condenser.

The OPERABILITY of the Ice Bed Temperature Monitoring System ensures that the capability is available for monitoring the ice bed temperature. The ice bed temperature may be determined at the local ice condenser temperature monitoring panel as well as in the Main Control Room and the Monitoring System would still be considered OPERABLE. In the event the Monitoring System is inoperable, the Required Actions provide assurance that the ice bed heat removal capacity will be retained within the specified time limits.

APPLICABILITY

The Ice Bed Temperature Monitoring System is required to be OPERABLE in MODES 1, 2, 3, and 4. This corresponds to the Applicability requirements for the ice bed in Technical Specification LCO 3.6.11, Ice Bed.

ACTIONS

A.1

With the ice bed temperature not available in the Main Control Room, the ice bed temperature must be determined at the local ice condenser temperature monitoring panel (local panel) every 12 hours. Since the ice bed has a very large thermal capacity, postulated off-normal conditions can be successfully tolerated for one or two weeks. Therefore, a 12 hour surveillance of the ice bed temperature will give sufficient warning of any incipient ice condenser temperature anomalies.

B.1.1, B.1.2, and B.1.3

With the Ice Bed Temperature Monitoring System inoperable and being unable to determine the ice bed temperature at the local panel, B.1.1, B.1.2, and B.1.3 require verification that: the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed; the last recorded mean ice bed temperature was less than or equal to 20°F (value does not account for instrument error) and steady; and the Ice Condenser Cooling System is OPERABLE.

(continued)

BASES

ACTIONS

B.1.1, B.1.2, and B.1.3 (continued)

The Completion Time of 1 hour and every 12 hours thereafter to perform Required Actions B.1.1 and B.1.3 and 1 hour to perform Required Action B.1.2 is reasonable and based upon the typical time necessary to perform the Required Actions. These Required Actions, along with the high thermal capacity of the ice bed, ensure that the ice bed will remain below critical temperatures while the Monitoring System is inoperable.

B.2.1 and B.2.2

With the Ice Bed Temperature Monitoring System inoperable and being unable to determine the ice bed temperature at the local panel, either the Monitoring System or the local monitoring panel must be restored to OPERABLE status within 30 days. A Completion Time of 30 days is given, provided that Required Actions B.1.1, B.1.2, and B.1.3 are met. These Required Actions, along with the high thermal capacity of the ice bed, ensure that the ice bed will remain below critical temperatures during the 30 day Completion Time. Also, the six alarmed temperature switches (which provide an alarm at 25°F) will continue to monitor the ice bed temperature. If the Ice Condenser Cooling System becomes inoperable before the Ice Bed Temperature Monitoring System is OPERABLE, then Required Action C must be performed.

C.1.1 and C.1.2

With the Ice Bed Temperature Monitoring System inoperable and being unable to determine the ice bed temperature at the local panel and with the Ice Condenser Cooling System not satisfying the minimum components OPERABILITY requirements of Required Action B.1.3, Required Actions C.1.1 and C.1.2 require verification that: the ice compartment lower inlet doors, intermediate deck doors, and top deck doors are closed; and that the last recorded mean ice bed temperature was less than or equal to 15°F (value does not account for instrument error) and steady. The Completion Time of 1 hour and every 12 hours thereafter to perform Required Action C.1.1 and 1 hour to perform Required Action C.1.2 is reasonable and based upon the typical time necessary to perform the Required Actions. These Required Actions, along with the high thermal capacity of the ice bed, ensure that the ice bed will remain below critical temperatures while the Monitoring System and Ice Condenser Cooling System are inoperable.

(continued)

BASES

ACTIONS (continued)

C.2.1, C.2.2, and C.2.3

With the Ice Bed Temperature Monitoring System inoperable and being unable to determine the ice bed temperature at the local panel and with the Ice Condenser Cooling System not satisfying the minimum components OPERABILITY requirements of Required Action B.1.3, the Ice Condenser Cooling System, Ice Bed Temperature Monitoring System or the local temperature monitoring panel must be restored to OPERABLE status. A Completion Time of 6 days is given, provided that Required Actions C.1.1, and C.1.2 are met. These Required Actions, along with the high thermal capacity of the ice bed, ensure that the ice bed will remain below critical temperatures during the 6-day Completion Time. Also, the six alarmed temperature switches (which provide an alarm at 25°F) will continue to monitor the ice bed temperature.

D.1 and D.2

If the Required Action and associated Completion Time of Condition A, B or C is not met, the integrity of the ice bed may be threatened. Therefore, the plant must be placed in a MODE in which the TR does not apply. This is done by placing the plant in MODE 3 in 6 hours and MODE 5 in 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.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.6.1.1

Performance of a CHANNEL CHECK on the Ice Bed Temperature Monitoring System once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of even something more serious. The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, TSR 3.6.1.1 ensures that loss of function will be identified within 12 hours.

(continued)

BASES (continued)

TR

The Inlet Door Position Monitoring System provides the only direct means of determining that the inlet doors are shut. Since an open door would allow heat input that could cause sublimation and mass transfer of ice in the ice condenser compartment, the Inlet Door Position Monitoring System must be OPERABLE whenever the ice bed is required to be OPERABLE. This ensures early detection of an inadvertently opened or failed door, allowing prompt action before ice bed degradation can occur.

APPLICABILITY

The Inlet Door Position Monitoring System is required to be OPERABLE in MODES 1, 2, 3 and 4. This corresponds to the Applicability requirements for the ice bed.

ACTIONS

A.1 and A.2

If the Inlet Door Position Monitoring System is inoperable in MODE 1, an alternate OPERABLE monitoring system must be used to ensure that the ice condenser is not degraded. This is done by confirming the Ice Bed Temperature Monitoring System is OPERABLE with the ice bed temperature ≤ 27°F (value does not account for instrument error). This Action must be completed within 4 hours and each 4 hours thereafter. The Frequency of 4 hours is based on the fact that temperature changes cannot occur rapidly in the ice bed because of the large mass of ice involved. Since this is an indirect means of monitoring inlet door position, operation in MODE 1 may continue for a maximum of 14 days in this condition. If the ice bed temperature increases to above 27°F, the ice bed must be declared inoperable in accordance with Technical Specification 3.6.11, "Ice Bed".

B.1

If the Required Action and associated Completion Time for Condition A are not met or if the Inlet Door Position Monitoring System is inoperable in MODES 2, 3, or 4, the Inlet Door Position Monitoring System must be restored to OPERABLE status within 48 hours. The 48-hour Completion Time is based on the fact that, with the very large mass of ice involved, it would not be possible for the temperature to increase to the melting point and a significant amount of ice to melt in a 48-hour period.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Action and associated Completion Time of Condition B cannot be met, the plant must be placed in a condition where OPERABILITY of the Inlet Door Position Monitoring System is not required. This is accomplished by placing the plant in MODE 4 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.6.2.1

Performance of the CHANNEL CHECK for the Inlet Door Position Monitoring System once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Performance of the CHANNEL CHECK helps to ensure that the instrumentation continues to operate properly between each TADOT. The dual switch arrangement on each door allows comparison of open and shut indicators for each zone as well as a check with the annunciator window. An alternate to the use of the annunciator window as the channel check, is to perform a continuity check of the same circuit used by the annunciator window. This continuity check will confirm if one or more inlet door zone switch contacts are closed which would represent an open inlet door. The Surveillance Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Thus, TSR 3.6.2.1 ensures that loss of function will be identified within 12 hours.

TSR 3.6.2.2

TSR 3.6.2.2 is the performance of a TADOT every 18 months. It checks trip devices (limit switches) that provide actuation signals directly. The 18-month Frequency was developed considering the plant conditions needed to perform TSR 3.6.2.2. The 18-month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.1.1

TSR 3.7.1.1 verifies that the pressures on the primary and the secondary sides in the steam generators are less than 200 psig (see Note below). At temperatures below 70° (see Note below), the temperature margin to RT_{NDT} is diminished. Hence, the pressure must be checked every hour to ensure that the material toughness criteria are not violated. The 1-hour Frequency is based on engineering judgment and is consistent with industry practice.

Note:
Instrument uncertainty has been considered in establishing these values and is discussed in this Bases section under Background.

REFERENCES

1. WCAP-13146, "Technical Basis for Determination of Secondary Side Pressure Test Temperatures in Sequoyah and Watts Bar Unit 1 and 2 Steam Generators."
2. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."
3. ASME Boiler and Pressure Code, Section III.
4. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.7 PLANT SYSTEMS

B 3.7.2 Flood Protection Plan

BASES

BACKGROUND

Nuclear power plants are designed to prevent the loss of capability for cold shutdown and maintenance thereof resulting from the most severe flood conditions that can reasonably be predicted to occur at the site as a result of severe hydrometeorological conditions, seismic activity, or both (Ref. 1). Assurance that safety-related facilities are capable of surviving all possible flood conditions is provided by the flood protection plan.

The elevations of plant features which could be affected by the submergence during floods vary from 714.5 ft Mean Sea Level (MSL) (access to electrical conduits) to 736.9 ft MSL (including wave runup). Plant grade is elevation 728 ft MSL which can be exceeded by extreme rainfall floods and closely approached by seismic-caused dam failure floods. A warning plan is needed to assure plant safety from floods.

The warning plan is divided into two stages. This two-stage plan is designed to allow adequate time for preparing the plant for operation in the flood mode and to avoid excessive economic loss in case a potential flood does not fully develop. Stage I warning, which is a minimum of 10 hours, allows preparation steps, causing some damage to be sustained, but will postpone major economic damage. Stage II warning, which is a minimum of 17 hours, is a warning that a forthcoming flood above grade is predicted.

Stage I procedures consist of a controlled reactor shutdown and other easily revokable steps, such as moving flood supplies above the probable maximum flood elevation and making temporary connections and load adjustments on the onsite power supply. After unit shutdown, the Reactor Coolant System will be cooled and the pressure will be reduced to less than 350 psig. Stage II procedures are the least easily revokable and more damaging steps necessary to have the plant in the flood mode when the flood exceeds plant grade. Heat removal from the steam generators will be accomplished by adding river water from the Fire Protection

(continued)

BASES

BACKGROUND (continued)	Due to valve design, ambient temperatures can affect the setpoints of the main steam safety valves (MSSVs), whereby a decrease in valve body temperature causes an increase in setpoint, resulting in non-conservative relief pressure. Ambient temperatures are monitored within the main steam valve vaults to ensure that the MSSVs minimum temperatures are maintained to meet the 1% code allowable variance on setpoints. Detailed BASES for the MSSVs is provided in Technical Specification B 3.7.1. The general guidelines, which are followed for the qualification of electrical equipment, are provided in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 1). Detailed requirements for the implementation of the general guidelines are provided in various Regulatory Guides and IEEE Standards. Basic requirements for the qualification of mechanical equipment are outlined in General Design Criteria 4 (Ref. 2).
APPLICABLE SAFETY ANALYSIS	Certain components, which have the service temperatures controlled by this requirement, are part of the primary success path and function to mitigate DBAs and transients. However, the integrity/OPERABILITY of these components is addressed in the relevant specifications that cover individual components. The service temperatures and the thermal aging, which are controlled by observing the requirements of this TR, are not inputs to the safety analysis. Further, Probabilistic Risk Assessment studies performed to date, do not explicitly model the function of area temperature monitors. In addition, this particular requirement covers only service temperatures and thermal aging of these components, which are not considerations in designing the accident sequences for theoretical hazard evaluations (Ref. 3).
TR	TR 3.7.5 provides nominal temperature limits in the vicinity of major equipment. The TR allows for each area shown in Table 3.7.5-1 to be higher or lower than the normal limit for a maximum of eight hours. Note that the temperature values listed in Table 3.7.5-1 do not account for instrument error.
APPLICABILITY	The limits on temperature and time apply whenever the affected equipment in an affected area is required to be OPERABLE.

(continued)

BASES (continued)

ACTIONS

A.1

Whenever the temperature in one or more areas have exceeded the normal temperature limits for more than eight hours, a report must be prepared and submitted to the NRC within 30 days. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation, and prepare the report.

Condition A has been modified by a Note stating that the provisions of TR 3.0.3 and TR 3.0.4 do not apply.

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

Whenever the temperature in one or more areas exceed the abnormal temperature limits, the temperature must be restored to within the normal limits in 4 hours. The Completion Time of 4 hours is based on operator experience and is a reasonable time for restoring the temperature. Alternatively, the affected equipment must be declared inoperable. Within 30 days a report must be prepared and submitted to NRC. The report must contain the cumulative time and the amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation and prepare the report.

C.1 and C.2

Whenever the temperature in the Intake Pumping Station mechanical or electrical equipment rooms exceeds the lower limit of 40 °F, actions must be initiated within 24 hours to ensure the temperature does not decrease below 32 °F. The Completion Time of 24 hours is based on temperature analysis. Within 7 days, restore normal temperatures within the areas affected. The 7 day Completion Time is based on a reasonable repair duration, and compensatory actions available during the interim period to maintain temperatures above 32 °F.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the temperature in the Intake Pumping Station mechanical or electrical equipment rooms decrease to 32 °F or lower, the affected equipment must be immediately declared inoperable. The Completion Time is based on potential freezing of safety-related components. Within 30 days a report must be prepared and submitted to NRC. The report must contain the cumulative time and amount by which the temperature has exceeded the limits. In addition, an analysis shall be submitted which demonstrates OPERABILITY of the affected equipment. The 30 day Completion Time is based on engineering experience and is a reasonable time to collect data, perform the evaluation and prepare the report.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.5.1

The temperatures for the areas listed in Table 3.7.5-1 must be determined every 12 hours to ensure compliance with the limits. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures. Note that the temperature values listed in Table 3.7.5-1 do not account for instrument error.

REFERENCES

1. 10 CFR 50.49 "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

2. 10 CFR 50 Appendix A, General Design Criteria 4, "Environmental and Dynamic Effects Design Bases."

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