ENCLOSURE 1

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WBN TECHNICAL SPECIFICATIONS

9802170151 980209
PDR ADOCK 05000390
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Acronym	Title
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARO	All Rods Out
ARFS	Air Return Fan System
ARV	Atmospheric Relief Valve
BOC	Beginning of Cycle
CAOC	Constant Axial Offset Control
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
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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AMENDMENT LISTING

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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.

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

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

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LIST OF EFFECTIVE PAGES

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TECHNICAL SPECIFICATIONS BASES

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TECHNICAL SPECIFICATIONS BASES

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TECHNICAL SPECIFICATIONS BASES

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TECHNICAL SPECIFICATIONS BASES

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REVISION LISTING

REVISIONS	ISSUED
NPF-20 Low Power Operating License	11-09-95
Revision 1	12-08-95
NPF-90 Full Power Operating License	02-07-96
Revision 2 (Amendment 1)	02-28-96
Revision 3	03-27-96
Revision 4 (Amendment 2)	06-13-96
Revision 5	07-03-96
Revision 6 (Amendment 3)	09-09-96
Revision 7	09-28-96
Revision 8	11-21-96
Revision 9	04-29-97
Revision 10 (Amendment 5)	05-27-97
Revision 11 (Amendment 6)	07-28-97
Revision 12	09-10-97
Revision 13 (Amendment 7)	09-11-97
Revision 14	10-10-97

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 (continued)

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

REFERENCES

1. Watts Bar FSAR, Section 6.0, "Engineered Safety Features."

2. Watts Bar FSAR, Section 7.0, "Instrumentation and Controls."

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

4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.

5. 10 CFR Part 50.49, "Environmental Qualifications of Electric Equipment Important to Safety for Nuclear Power Plants."

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

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

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

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

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

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

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

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

- Field transmitters or process sensors: provide a measurable electronic signal or contact actuation based on the physical characteristics of the parameter being measured;

- Signal processing equipment including process protection system, and field contacts: provide analog to digital conversion (Digital Protection System), signal conditioning, setpoint comparison, process algorithm actuation (Digital Protection System), compatible electrical signal output to protection system devices, and control board/control room/ miscellaneous indications; and

- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable, setpoint comparators, or contact outputs from the signal process control and protection system.

Field Transmitters or Sensors

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Containment Isolation (continued)

leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since the CCS is required to support RCP operation, not isolating the CCS on the low pressure Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating the CCS on the low pressure signal may force the use of feed and bleed cooling, which could prove more difficult to control.

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of the CCS and ERCW, are isolated. CCS is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and ERCW to air or oil coolers. All process lines not equipped with remote operated isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room or from local panel(s). Either switch actuates both trains. Note that manual actuation of Phase A Containment Isolation also actuates Containment Vent Isolation.

The Phase B signal isolates the CCS. This occurs at a relatively high containment pressure that is indicative of a large break LOCA or an SLB. For these events, forced circulation using the RCPs is no longer desirable. Isolating the CCS at the higher pressure does not pose a challenge to the containment boundary because the CCS is a closed loop inside containment. Although some system components do not meet all of the ASME Code requirements applied to the containment itself, the system is continuously pressurized to a pressure greater than the Phase B setpoint. Thus, routine operation demonstrates the integrity of the system pressure boundary for pressures exceeding the Phase B setpoint. Furthermore, because system

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Containment Isolation (continued)

pressure exceeds the Phase B setpoint, any system leakage prior to initiation of Phase B isolation would be into containment. Therefore, the combination of the CCS design and Phase B isolation ensures the CCS is not a potential path for radioactive release from containment.

Phase B containment isolation is actuated by Containment Pressure–High High, or manually, via the automatic actuation logic, as previously discussed. For containment pressure to reach a value high enough to actuate Containment Pressure–High High, a large break LOCA or SLB must have occurred and containment spray must have been actuated. RCP operation will no longer be required and CCS to the RCPs is, therefore, no longer necessary. The RCPs can be operated with seal injection flow alone and without CCS flow to the thermal barrier heat exchanger.

Manual Phase B Containment Isolation is accomplished by the same switches that actuate Containment Spray. When the two switches in either set are turned simultaneously, Phase B Containment Isolation and Containment Spray will be actuated in both trains.

a. Containment Isolation-Phase A Isolation

(1) Phase A Isolation-Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room or from local panel(s). Either switch actuates both trains. Note that manual initiation of Phase A Containment Isolation also actuates Containment Vent Isolation.

(2) Phase A Isolation-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.7

SR 3.3.2.7 is the performance of a SLAVE RELAY TEST for slave relays K603A, K603B, K604A, K604B, K607A, K607B, K609A, K609B, K612A, K625A, and K625B. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment which may be operated in the design mitigation MODE is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment which may not be operated in the design mitigation MODE is prevented from operation by the slave relay test circuit.

For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 18 months. The Frequency is justified by TVA correspondence to the NRC dated November 9, 1984 (Ref. 9) and Design Change Notice W-38238-A associated documentation (Reference 12), and for relays K607A, K607B, and K612A, Westinghouse letter to TVA (Ref. 11).

SR 3.3.2.8

SR 3.3.2.8 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and AFW pump start on trip of all MFW pumps. It is performed every 18 months. The Frequency is based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation functions. The manual initiation functions have no associated setpoints.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

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

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the Watts Bar setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.9 (continued)

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

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

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

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in Technical Requirements Manual, Section 3.3.2 (Ref. 8). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

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.

ESF RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.10 (continued)

Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 1092 psig in the SGs.

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per RTB cycle. This Frequency is based on operating experience demonstrating that undetected failure of the P-4 interlock sometimes occurs when the RTB is cycled.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

REFERENCES

1. Watts Bar FSAR, Section 6.0, "Engineered Safety Features."

2. Watts Bar FSAR, Section 7.0, "Instrumentation and Controls."

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

4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.

5. Code of Federal Regulations, Title 10, Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants."

(continued)

BASES

REFERENCES (continued)

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

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

8. Watts Bar Technical Requirements Manual, Section 3.3.2, "Engineered Safety Feature Response Times."

9. TVA Letter to NRC, November 9, 1984, "Request for Exemption of Quarterly Slave Relay Testing, (L44 841109 808)."

10. Evaluation of the applicability of WCAP-10271-P-A, Supplement 1, and Supplement 2, Revision 1, to Watts Bar.

11. Westinghouse letter to TVA (WAT-D-8347), September 25, 1990, "Charging/Letdown Isolation Transients" (T33 911231 810).

R1 12. Design Change Notice W-38238 associated documentation.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

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

1. Manual Initiation

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

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

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

2. Automatic Actuation Logic and Actuation Relays

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

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable

(continued)

BASES

LCO

2. Automatic Actuation Logic and Actuation Relays (continued)

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

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Vent Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups and sample pump operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

Only the Allowable Value is specified for the Containment Purge Exhaust Radiation Monitors in the LCO. The Allowable Value is based on expected concentrations for a small break LOCA, which is more restrictive than 10 CFR 100 limits. The Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip function. The actual nominal Trip Setpoint is normally still more conservative than that required by the Allowable Value. If the setpoint does not exceed the Allowable Value, the radiation monitor is considered OPERABLE.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.5 (continued)

block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.6

The required RHR suction relief valve shall be demonstrated OPERABLE by verifying both RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.

Every 31 days both RHR suction isolation valves are verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valves must be locally verified in the open position with the manual actuator locked. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.

SR 3.4.12.7

The COT is required to be in frequency prior to decreasing RCS temperature to ≤ 350°F or be performed within 12 hours after decreasing RCS temperature to ≤ 350°F on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required. The COT is required to be performed every 31 days when RCS temperature is ≤ 350°F with the reactor head in place.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.7 (continued)

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

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

SR 3.4.12.8

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

REFERENCES

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

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

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

4. Watts Bar FSAR, Section 15.2, "Condition II - Faults of Moderate Frequency."

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

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

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

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

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

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for particulate monitoring and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

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

(continued)

BASES (continued)

BACKGROUND (continued)

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

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

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

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

APPLICABLE SAFETY ANALYSES

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

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak detrimental to the safety of the unit and the public occur. RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

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

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

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

APPLICABILITY

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

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

BASES (continued)

ACTIONS

A.1 and A.2

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

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

Required Action A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment pocket sump level monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

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

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

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

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

BASES (continued)

ACTIONS

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

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

Required Action B.1 and Required Action B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitor channel is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

C.1 and C.2

If a Required Action of Condition A or B cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.2

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

SR 3.4.15.3 and SR 3.4.15.4

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

REFERENCES

1. 10 CFR 50, Appendix A, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."

2. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," Revision 0, May 1973.

3. Watts Bar FSAR, Section 5.2.7, "RCPB Leakage Detection Systems."

BASES

APPLICABLE SAFETY ANALYSES (continued)

As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated solely by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and centrifugal charging pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease until they are not required and the centrifugal charging pumps become solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 3) will be met following a LOCA:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty. For large breaks, an increase in water volume can be either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of 7627 gallons and 8082 gallons. To allow for instrument inaccuracy, values of 7717 gallons and 8004 gallons are specified.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure analysis limit of 690 psig prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity. The LOCA analysis assumes a value of 585 psig. To account for the accumulator tank design pressure rating, and to allow for instrument accuracy values of $\geq$ 610 psig and $\leq$ 660 psig are specified for the pressure indicator in the main control room.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

(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.

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1 (continued)

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, Option B, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."

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

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

4. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, 8 ft 7 inches in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the plant safety analyses.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment air lock door is opened, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door status available to operations personnel and because the interlock is only disabled in MODES 5 and 6. R5

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, Option B, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal or which are normally closed. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the purge and exhaust valves receive an isolation signal on a containment high radiation condition. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the

(continued)

BASES

LCO (continued)

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

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

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

APPLICABILITY

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

ACTIONS

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

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

(continued)

BASES

ACTIONS (continued)

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

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

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

A.1 and A.2

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

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

(continued)

ENCLOSURE 3

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WBN TECHNICAL REQUIREMENTS MANUAL

TECHNICAL REQUIREMENTS

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Core Operating Limits Report, Revision 1

Process Control Program, Revision 1

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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B 3.6-11	0	09/30/95
B 3.6-12	0	09/30/95
B 3.7-1	0	09/30/95
B 3.7-2	0	09/30/95
B 3.7-3	0	09/30/95
B 3.7-4	0	09/30/95
B 3.7-5	0	09/30/95
B 3.7-6	0	09/30/95

LIST OF EFFECTIVE PAGES

PAGE	REVISION	DATE
B 3.7-7	0	09/30/95
B 3.7-8	0	09/30/95
B 3.7-9	0	09/30/95
B 3.7-10	0	09/30/95
B 3.7-11	0	09/30/95
B 3.7-12	0	09/30/95
B 3.7-13	5	08/29/97
B 3.7-14	5	08/29/97
B 3.7-15	4	08/18/97
B 3.7-16	5	08/29/97
B 3.7-17	5	08/29/97
B 3.7-18	0	09/30/95
B 3.7-19	0	09/30/95
B 3.7-20	0	09/30/95
B 3.7-21	0	09/30/95
B 3.7-22	0	09/30/95
B 3.7-23	2	01/04/96
B 3.7-24	0	09/30/95
B 3.7-25	0	09/30/95
B 3.8-1	0	09/30/95
B 3.8-2	0	09/30/95
B 3.8-3	0	09/30/95
B 3.8-4	0	09/30/95
B 3.8-5	0	09/30/95
B 3.8-6	0	09/30/95
B 3.8-7	0	09/30/95
B 3.8-8	0	09/30/95
B 3.8-9	0	09/30/95
B 3.8-10	0	09/30/95
B 3.8-11	0	09/30/95
B 3.8-12	0	09/30/95
B 3.8-13	0	09/30/95
B 3.8-14	0	09/30/95
B 3.8-15	0	09/30/95
B 3.8-16	0	09/30/95
B 3.8-17	0	09/30/95
B 3.8-18	0	09/30/95
B 3.8-19	0	09/30/95
B 3.8-20	0	09/30/95
B 3.8-21	0	09/30/95
B 3.8-22	0	09/30/95
B 3.9-1	0	09/30/95
B 3.9-2	0	09/30/95
B 3.9-3	0	09/30/95
B 3.9-4	0	09/30/95
B 3.9-5	0	09/30/95
B 3.9-6	0	09/30/95
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B 3.9-8	0	09/30/95
B 3.9-9	0	09/30/95

TECHNICAL REQUIREMENTS MANUAL

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REVISION LISTING

Revisions	Issued
Revision 0	09-30-95
Revision 1	12-06-95
Revision 2	01-04-96
Revision 3	02-28-96
Revision 4	08-18-97
Revision 5	08-29-97
Revision 6	09-08-97
Revision 7	09-12-97
Revision 8	09-22-97
Revision 9	10-10-97

TECHNICAL SURVEILLANCE REQUIREMENTS

----------NOTES----------

1. 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.

2. TSR 3.1.5.4, TSR 3.1.5.5 and TSR 3.1.5.6 are only required to be performed if the Boric Acid Storage System is the required borated water source.

SURVEILLANCE		FREQUENCY
TSR 3.1.5.1	----------NOTE---------- Only required when ambient air temperature is < 60 °F. ---------- Verify RWST solution temperature is ≥ 60 °F.	24 hours
TSR 3.1.5.2	Verify RWST boron concentration is ≥ 2,500 ppm.	7 days
TSR 3.1.5.3	Verify RWST borated water volume is ≥ 62,900 gallons.	7 days
TSR 3.1.5.4	Verify Boric Acid Tank (BAT) solution temperature is ≥ 63°F.	24 hours
TSR 3.1.5.5	Verify BAT boron concentration is ≥ 6,120 and ≤ 6,990 ppm.	7 days
TSR 3.1.5.6	Verify BAT borated water volume is ≥ 3,800 gallons.	7 days

TR 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.6 Borated Water Sources, Operating

TR 3.1.6 The following borated water sources shall be OPERABLE as required by TR 3.1.2:

a. A Boric Acid Storage System, and

b. The Refueling Water Storage Tank (RWST).

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required Boric Acid Storage System, inoperable.	A.1 Restore Boric Acid Storage System, to OPERABLE status.	72 hours
	OR	
	A.2.1 Be in MODE 3.	78 hours
	AND	
	A.2.2 Borate to a SDM equivalent to ≥ 1% Δk/k at 200°F.	78 hours
	AND	
	A.2.3 Restore Boric Acid Storage System to OPERABLE status.	246 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	C.1 Restore RWST to OPERABLE status.	8 hours
D. RWST inoperable for reasons other than Condition C.	D.1 Restore RWST to OPERABLE status.	1 hour
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3 AND E.2 Be in MODE 4 with one or more RCS cold leg temperatures ≤ 310 °F.	6 hours 12 hours

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.6.1	----------NOTE---------- Only required when outside air temperature is < 60 °F or >105 °F. ---------- Verify RWST solution temperature is ≥ 60 °F and ≤ 105 °F.	24 hours
TSR 3.1.6.2	Verify RWST boron concentration is ≥ 2,500 ppm and ≤ 2,700 ppm.	7 days

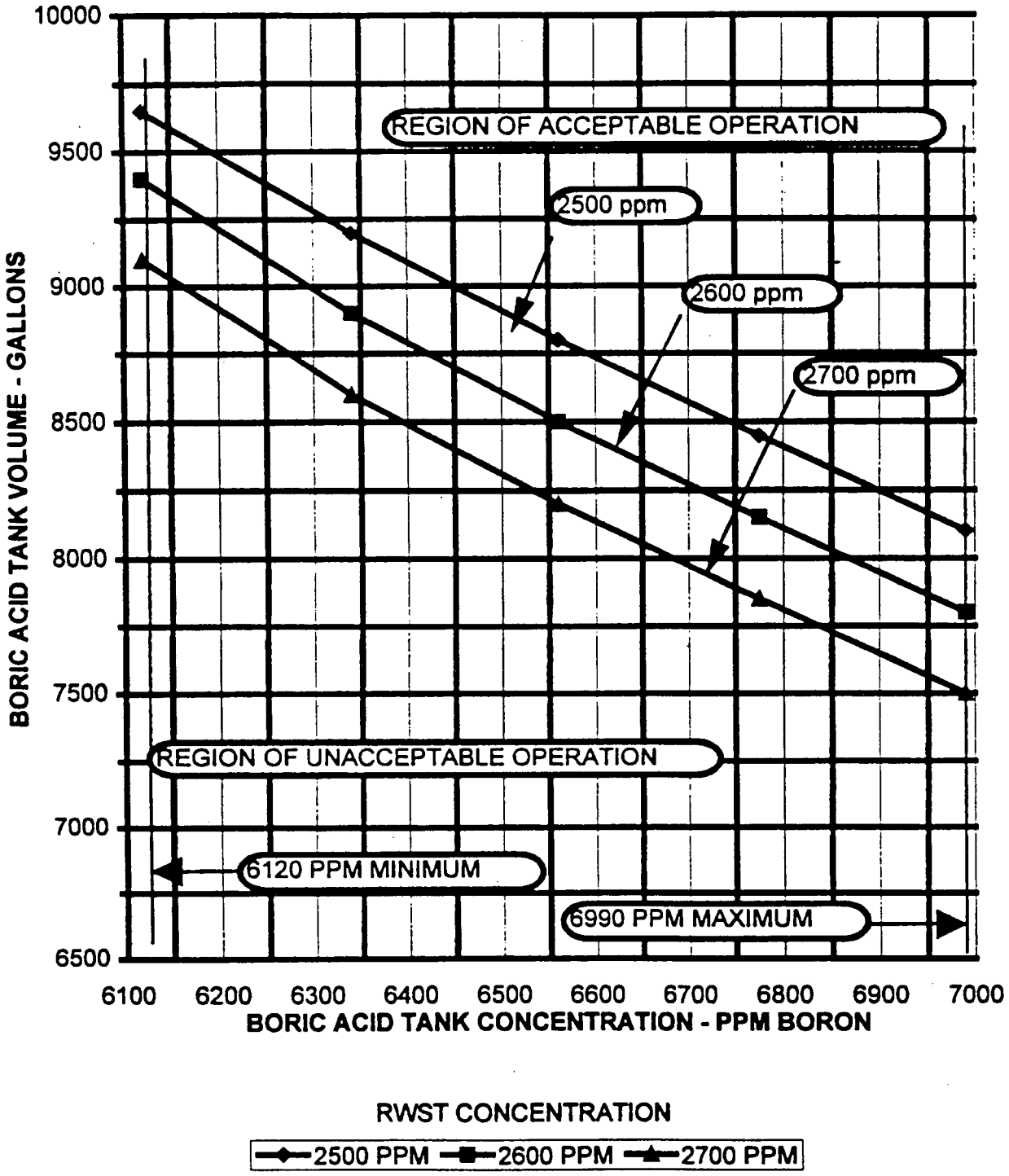
(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.1.6.3	Verifiy RWST borated water volume is ≥ 370,000 gallons.	7 days
TSR 3.1.6.4	----------NOTE---------- Only required if the BAT is required OPERABLE. ---------- Verify Boric Acid Tank (BAT) solution temperature is ≥ 63°F.	24 hours
TSR 3.1.6.5	----------NOTE---------- Only required if the BAT is required OPERABLE. ---------- Verify BAT boron concentration is in accordance with Figure 3.1.6.	7 days
TSR 3.1.6.6	----------NOTE---------- Only required if the BAT is required OPERABLE. ---------- Verify BAT borated water volume is in accordance with Figure 3.1.6.	7 days

TECHNICAL REQUIREMENT FIGURE 3.1.6

BORIC ACID TANK LIMITS
BASED ON RWST BORON CONCENTRATION



TR 3.1 REACTIVITY CONTROL SYSTEMS

TR 3.1.7 Position Indication System, Shutdown

TR 3.1.7 The group demand position indicators shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown or control rod that is not fully inserted.

APPLICABILITY: MODES 3, 4, and 5, when the reactor trip breakers are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more group demand position indicators inoperable.	A.1 Open reactor trip breakers.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.1.7.1	Determine that each group demand position indicator is OPERABLE by movement of the associated shutdown or control rod 10 steps in any one direction.	Within 4 hours after closing the reactor trip breakers if not completed within previous 31 days. AND 31 days thereafter

Table 3.3.1-1 (Page 1 of 2)

Reactor Trip System Instrumentation Response Times

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux	
a. High	≤ 0.5 second[1]
b. Low	≤ 0.5 second[1]
3. Power Range, Neutron Flux	
a. High Positive Rate	N.A.
b. High Negative Rate	N.A.
4. Intermediate Range, Neutron Flux	N.A.
5. Source Range, Neutron Flux	N.A.
6. Overtemperature ΔT	≤ 8 seconds[1]
7. Overpower ΔT	≤ 8 seconds[1]
8. Pressurizer Pressure	
a. Low	≤ 2 seconds
b. High	≤ 2 seconds
9. Pressurizer Water Level--High	N.A.

(continued)

(1) Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from the detector output or input of first electronic component in channel.

Table 3.3.1-1 (Page 2 of 2)

Reactor Trip System Instrumentation Response Times

FUNCTIONAL UNIT	RESPONSE TIME
10. Reactor Coolant Flow - Low	
a. Single Loop (Above P-8)	≤ 1.2 seconds
b. Two Loops (Above P-7 and Below P-8)	≤ 1.2 seconds
11. Undervoltage-Reactor Coolant Pumps	≤ 1.5 seconds[2]
12. Underfrequency-Reactor Coolant Pumps	≤ 0.6 second[3]
13. Steam Generator Water Level-Low-Low	≤ 2 seconds[4]
14. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
15. Safety Injection Input from ESF	N.A.
16. Reactor Trip System Interlocks	N.A.
17. Reactor Trip Breakers	N.A.
18. Reactor Trip Breaker UV and ST	N.A.
19. Automatic Trip and Interlock Logic	N.A.

(2) Includes sensor delay time, adjustable time delay, logic and breaker trip times, gripper release (150 msec.) and EMF decay time (250 msec.).

(3) Includes sensor delay time, adjustable time delay, logic and breaker trip times and gripper release time (150 msec.).

(4) With Trip Time Delay (TTD) = 0 seconds.

Table 3.3.2-1 (Page 3 of 5)

Engineered Safety Features Actuation System Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Steam Generator Water Level - Low - Low	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 60[7]
b. Turbine-driven Auxiliary Feedwater Pumps	≤ 60[8]
9. DELETED	
10. RWST Level-Low Coincident with Containment Sump Level-High and Safety Injection	
Automatic Switchover to Containment Sump	≤ 250
11. Loss-of-Offsite Power	
Auxiliary Feedwater Pumps	≤ 60
12. Trip of All Main Feedwater Pumps	
Auxiliary Feedwater Pumps	≤ 60
13. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	
a. Motor-driven Auxiliary Feedwater Pumps	≤ 47
b. Turbine-driven Auxiliary Feedwater Pumps	≤ 42
14. Loss of Voltage/Degraded Voltage	
6.9 kV Shutdown Board	≤ 12[9]
15. MSV Vault Room Water Level - High	
a. North MSV Vault Room	≤ 8.5[15]
b. South MSV Vault Room	≤ 8.5[15]

Table 3.3.2-1 (Page 4 of 5)

Engineered Safety Features Actuation System Response Times

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting and sequence loading delay not included. Offsite power available.

(3) Air operated valves.

(4) Offsite power available - diesel generator starting and sequence loading delays not included. Response time limit includes the opening of valves to establish flowpath and bringing the pumps to full speed. The additional sequential transfer of CCP suction from the VCT to the RWST (RWST valves open, then the VCT valves close) is included.

(5) Diesel generator starting and sequence loading delays included. Response time limit includes the opening of valves to establish flow path and bringing the pumps up to full speed. The additional sequential transfer of suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(6) The following equipment are exceptions to the response time shown in the table and will have the following response times for the initiating signals and functions:

A. Fire Protection CIVs	22(2)/32(1)
B. Ice Condenser CIVs	32
C. Excess Letdown Hx Supply CIV	68(2)/78(1)
D. EGTS Fans	20(2)/30(1)
E. Required for EGTS OPERABILITY	
1. Fire Protection Secondary CIVs	20(2)/30(1)
2. Secondary Containment Purge Isolation Valves	12.7(2)/22.7(1)
F. Steam Generator Blowdown CIVs	17(2)/27(1)

(7) On 2/3 any steam generator and Trip Time Delay = 0 seconds.

(8) On 2/3 in 2/4 steam generators and Trip Time Delay = 0 seconds.

(9) The response time is measured from the time the 6.9 kV shutdown boards voltage exceeds the Setpoint until the time full voltage is returned for the loss of voltage sensors; or from the time the degraded voltage timers generate a signal to trip the feeder breakers and shed loads until the time full voltage is returned for the degraded voltage sensors.

(10) The Response Time for motor-driven AFW pumps includes the diesel generator starting and sequence loading delays. The Response Time for (steam) turbine driven AFW pumps does not include diesel generator starting and sequence loading delays.

(continued)

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]			
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-52-75E 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 Peak Accelerographs			
a. 0-XR-52-76A Reactor Bldg. El. 725	1	TSR 3.3.4.3	0 - 5.0 g
b. 0-XR-52-76B Reactor Bldg. El. 730	1	TSR 3.3.4.3	0 - 5.0 g
c. 0-XR-52-76D Control Bldg. El. 755	1	TSR 3.3.4.3	0 - 5.0 g
3. Triaxial Seismic Switches			
a. 0-XS-52-80 Annulus El. 703[1]	1	TSR 3.3.4.1 TSR 3.3.4.2[4] TSR 3.3.4.3	0.025 - 0.25 g
4. Triaxial Response-Spectrum Recorders			
a. 0-XR-52-77A Annulus El. 703	1	TSR 3.3.4.1 TSR 3.3.4.2[4] TSR 3.3.4.3	2 - 25.4 Hz
b. 0-XR-52-77B Reactor Bldg. El. 757	1	TSR 3.3.4.3	2 - 25.4 Hz
c. 0-XR-52-77D Aux. Cont. Rm. El. 757	1	TSR 3.3.4.3	2 - 25.4 Hz
d. 0-XR-52-77E D/G Bldg. El. 742	1	TSR 3.3.4.3	2 - 25.4 Hz

(1) With associated acceleration triggers, and control room indication on 0-XR-52-75.
(2) Except acceleration trigger.
(3) Includes acceleration trigger.
(4) Except setpoint verification.

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)

Table 3.7.3-4 (Page 2 of 2)

Snubber Functional Testing Plan

TABLE NOTES

1. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested.

2. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type.

3. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan.

4. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

5. Administrative controls are required for snubber removal for functional surveillance testing during plant operating Modes 1 through 4.

Table 3.7.3-5 (Page 1 of 1)

Snubber Functional Testing Acceptance Criteria

The snubber functional test shall verify that:

a. Activation (restraining action) is achieved within the specified range in both tension and compression.

b. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range (Hydraulic Snubbers).

c. The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel (Mechanical Snubbers).

TABLE NOTES

1. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

2. An engineering evaluation shall be made of each failure to meet the functional test criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of the type which may be subject to the same failure mode.

3. For snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Table 3.7.3-4 for snubbers not meeting the functional test acceptance criteria.

Table 3.7.5-1 (Page 1 of 2)
Area Temperature Monitoring

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

(continued)

Table 3.7.5-1 (Page 2 of 2)
Area Temperature Monitoring

AREA	NORMAL LIMIT °F	ABNORMAL LIMIT °F
19. South steam valve vault room U1. (at affected MSSVs)	≥ 50	≥ 50
20. Add Equip Bldg U1 el 729 between UHI accumulators.	≤ 92	≤ 110
21. CB Main Control Room south wall.	≤ 80	≤ 104
22. CB Main Control Room across from 1-M-9.	≤ 80	≤ 104
23. CB Computer room el 708 center of room.	≤ 74	≤ 104
24. CB Aux. Instrument Room el 708.	≤ 90	≤ 104
25. D/G Bldg el 742 2B-B D/G room on wall by battery charger.	≤ 104	≤ 120
26. D/G Bldg el 742 1A-A D/G Room near D/G set.	≥ 50	≥ 50
27. D/G Bldg el 742 1B-B D/G Room near D/G set.	≥ 50	≥ 50
28. D/G Bldg el 742 2A-A D/G Room near D/G set.	≥ 50	≥ 50
29. D/G Bldg el 742 2B-B D/G Room near D/G set.	≥ 50	≥ 50
30. D/G Bldg el 760.5 next to 480V diesel Aux Bd 2B1-B.	≤ 104	≤ 120
31. IPS Mechanical Equipment Room 1 el 722 near ERCW and HPFP Instruments and sense lines.	≥50≤104	≥40≤115
32. IPS Mechanical Equipment Room 2 el 722 near ERCW and HPFP Instruments and sense lines.	≥50≤104	≥40≤115
33. IPS el 741 in B train ERCW pump room.	≤ 120	≤ 120
34. IPS el 711 next to 480V IPS board and transformer (A bus).	≥50≤104	≥40≤115
35. IPS el 711 next to 480V IPS board and transformer (B bus).	≤ 104	≤ 115
36. Add D/G Bldg el 742 C-S D/G Room near D/G set.	≥ 50	≥ 50

Table 3.8.4-1

Submerged Components With Automatic De-energization
Under Accident Conditions

6.9kV SHUTDOWN BD 1A-A

COMPT	LOAD	FCTN
20	1-DPL-68-341A-A	SI
21	1-DPL-68-341F	SI

6.9kV SHUTDOWN BD 1B-B

COMPT	LOAD	FCTN
20	1-DPL-68-341D-B	SI
21	1-DPL-68-341H	SI

480V SHUTDOWN BD 1A1-A

COMPT	LOAD	FCTN
7B	1-MTR-30-83/1-A	CIB
7D	1-MTR-30-83/2-A	CIB
7C	1-MTR-30-74-A	CIB

480V SHUTDOWN BD 1B1-B

COMPT	LOAD	FCTN
7C	1-MTR-30-92/1-B	CIB
10D	1-MTR-30-92/2-B	CIB
7D	1-MTR-30-75-B	CIB

480V SHUTDOWN BD 1A2-A

COMPT	LOAD	FCTN
7A	1-MTR-30-88/1-A	CIB
8A	1-MTR-30-88/2-A	CIB
7D	1-MTR-30-77-A	CIB

480V SHUTDOWN BD 1B2-B

COMPT	LOAD	FCTN
7B	1-MTR-30-80/1-B	CIB
10C	1-MTR-30-80/2-B	CIB
7D	1-MTR-30-78-B	CIB

480V REACTOR MOV BD 1A1-A

COMPT	LOAD	FCTN
16A	1-MTR-31-265	CIA

480V REACTOR MOV BD 1B1-B

COMPT	LOAD	FCTN
16A	1-MTR-31-266	CIA
16E	1-PO-213-B1/(1-5)	SI
17E	1-PO-213-B1/(6-10)	SI

480V REACTOR VENT BD 1A-A

COMPT	LOAD	FCTN
2A	1-MTR-77-125A	SI
9B	1-MTR-30-95	CIB
10B	1-MTR-30-99	CIB
11D	1-MTR-77-4	CIA

480V REACTOR VENT BD 1B-B

COMPT	LOAD	FCTN
2A	1-MTR-77-125B	SI
9B	1-MTR-30-97	CIB
10B	1-MTR-30-100	CIB
11D	1-MTR-77-6	CIA

125VDC VITAL BATTERY BD I

CKT	LOAD	FCTN
A6	1-FCV-62-72-A	CIA
A7	1-FCV-62-73-A	CIA
A8	1-FCV-62-74-A	CIA
A17	1-FCV-62-76-A	CIA
A31	1-FCV-63-71-A	CIA
B30	1-FSV-30-56-A	CVI
B32	1-FSV-30-20-A	CVI
B36	1-FSV-30-40-A	CVI
C4	1-FCV-31-308-A	CIA
C22	1-FCV-1-181-A	CIA
C38	1-FCV-1-183-A	CIA

125VDC VITAL BATTERY BD II

CKT	LOAD	FCTN
A20	1-FCV-43-2-B	CIA
A21	1-FCV-43-11-B	CIA
A22	1-FCV-43-22-B	CIA
A24	1-FCV-77-16-B	CIA
A43	1-FCV-77-127-B	CIA
A44	1-FCV-77-9-B	CIA
A45	1-FCV-77-18-B	CIA
C18	1-FCV-30-8-B	CVI
C21	1-FCV-43-75-B	CIA
C26	1-FCV-61-122-B	CIA
C40	1-FCV-30-15-B	CVI
C41	1-FCV-43-54D-B	CIA
C42	1-FCV-43-56D-B	CIA
C43	1-FCV-43-59D-B	CIA
C44	1-FCV-43-63D-B	CIA

CIA - CONTAINMENT ISOLATION PHASE A
CIB - CONTAINMENT ISOLATION PHASE B
CVI - CONTAINMENT VENT ISOLATION
SI - SAFETY INJECTION

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. 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)

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 at least 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 at least 62,900 gallons. 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 at least 63°F. 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 at least 3,800 gallons. 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

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.6.2 (continued)

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. 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. 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 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.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

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. 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)

ACTIONS

A.1

With one or more group demand position indicators inoperable, the plant must be placed in a condition where the demand position indicators are not required. This is accomplished by opening the reactor trip breakers immediately.

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

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.7.1

Exercising rods at a Frequency of 31 days allows the operator to determine that all withdrawn rods, including the group step counter demand position indicator, continue to be OPERABLE. A movement of 10 steps is adequate to demonstrate motion and verify a corresponding step change in the group step counter demand position indicator. Four hours is provided to perform the first surveillance after closing the reactor trip breakers if the surveillance has not been performed within the previous 31 days. The 31-day Frequency takes into consideration other information available to the operator in the control room and the remote likelihood that rods would be withdrawn from fully inserted for extended periods of time during shutdown conditions.

REFERENCES

1. Watts Bar FSAR, Section 4.2.3 "Reactivity Control System."

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

BASES (continued)

APPLICABLE SAFETY ANALYSES	The Turbine Overspeed Protection System trips the turbine to prevent the generation of potentially damaging missiles from the turbine, in the event of a loss of the Turbine Speed Control System, or a transient. However, the turbine overspeed event is not a DBA (Ref. 2). Turbine Overspeed Protection is not assumed to function in the safety analyses.
TR	This requirement is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could present a personnel and equipment hazard.
APPLICABILITY	At least one Turbine Overspeed Protection System must be OPERABLE whenever the potential for turbine overspeed exists. Since steam may be admitted to the turbine in MODES 1, 2, or 3, the requirement is applicable in these MODES. The Applicability has been modified by a Note stating that it is 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. Under these conditions, the potential for turbine overspeed does not exist. Another Note has been added stating TR 3.0.4 is not applicable. Failure to meet this requirement does not prohibit MODE changes.
ACTIONS	<u>A.1.1 and A.2.1</u> If one high pressure turbine steam inlet valve is inoperable, action must be taken to verify the two high pressure turbine steam inlet valves on the same steam chest which are opposite the inoperable valve are OPERABLE. The verification of operability (by testing) is needed to assure that these valves will close when the turbine is tripped. The 6-hour Completion Time was developed taking into account the time to reduce power and conduct the functional testing of the turbine steam inlet valves. <u>A.1.2</u> If one high pressure turbine steam inlet valve is inoperable, action must be taken to restore the inoperable valve to OPERABLE status. The 72-hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE valves in the same steam chest and reasonable time for repairs. OPERABILITY is established by either having a fully functional valve or manually closing (continued)

BASES

ACTIONS

A.1.2 (continued)

the otherwise inoperable valve. Closing the inoperable valve ensures total steam isolation to the high pressure turbine in the event of an overspeed condition, even with a single failure of another valve. Since the turbine must not have flow into non-adjacent zones (i.e., governor valves 1 and 3 or governor valves 2 and 4) due to possible turbine damage, isolation of a steam chest is precluded.

A.2.2

This alternative ensures total steam isolation to the high pressure turbine in the event that the inoperable valve cannot be closed. If it is determined through testing that another valve is also inoperable, then sufficient time would be available for an orderly turbine shutdown. This action is intended to prevent a condition involving a turbine trip coincident with multiple inoperable high pressure turbine steam inlet valves resulting in steam flow to the turbine. In this condition, insufficient time would be available for the operator to isolate the high speed turbine to avoid turbine overspeed. Since the turbine must not have flow through non-adjacent zones (i.e., governor valves 1 and 3 or governor valves 2 and 4) due to possible turbine damage, isolation of a steam chest is precluded. An additional 6 hours (total of 78 hours) are allowed for a power reduction, if necessary, in an orderly manner and without challenging plant systems.

A.3

Another alternative is to isolate the turbine from the steam supply using the Main Steam Isolation Valves (MSIV). This alternative assumes there may be partial leakage through the high pressure turbine steam inlet valves, or the steam path into the turbine cannot be isolated by the turbine inlet valves, thereby requiring closure of the MSIVs. This places the turbine in a condition where overspeed protection is not required. Again, an additional 6 hours (total of 78 hours) are allowed for a power reduction, if necessary, in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS

B.1

If one reheat stop valve or one reheat intercept valve in one or more low pressure turbine steam lines is inoperable, action must be taken to restore the inoperable valve(s) to OPERABLE status. The 72-hour Completion Time was developed taking into account the redundant capabilities afforded by the OPERABLE valve in the same steam line(s) and reasonable time for repairs.

B.2

A first alternative to Required Action B.1 is to close at least one valve in the affected steam line(s). This places the low pressure steam line(s) with the inoperable valve(s) in a no flow condition. This ensures total steam isolation to the low pressure turbine(s) in the event of an overspeed condition, even with a single failure of another reheat stop valve or reheat intercept valve. An additional 6 hours (total of 78 hours) are allowed for a power reduction, if necessary, in an orderly manner and without challenging plant systems.

B.3

A second alternative is to isolate the turbine from the steam supply. This places the turbine in a condition where overspeed protection is not required. Again, an additional 6 hours (total of 78 hours) are allowed for a power reduction, if necessary, in an orderly manner and without challenging plant systems.

C.1

If the Turbine Overspeed Protection System is inoperable for causes other than Condition A or Condition B, the turbine must be placed in a condition where overspeed protection is not required. This is accomplished by isolating the turbine from the steam supply system. A Completion Time of 6 hours is allowed to shutdown the turbine in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.5.1

The Turbine Overspeed Protection System testing requirements and frequencies are provided in Reference 3. In addition, the Surveillance is modified by a Note stating that TSR 3.0.4 is not applicable.

REFERENCES

1. Watts Bar FSAR, Section 10.2, "Turbine Generator."

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

3. PAI-6.02, "Turbine Integrity Program With Turbine Overspeed Protection (TIPTOP)."

B 3.3 INSTRUMENTATION

B 3.3.6 Loose-Part Detection System

BASES

BACKGROUND

The Loose-Part Detection System consists of six sensors, a system cabinet, alarm units, a frequency-modulated tape recorder, an audio monitor, and calibration devices. The sensors are located in the six natural collection regions. These regions consist of the top and bottom plenums of the reactor vessel and the primary coolant inlet plenum to each steam generator. There are installed spares at each sensor location. The entire system is described in Reference 1.

The Loose-Part Detection System provides the capability to detect acoustic disturbances indicative of loose parts within the Reactor Coolant System (RCS) pressure boundary. This system is provided to avoid or mitigate damage to RCS components that could occur from these loose parts. The Loose-Part Detection System Technical Requirement is consistent with the recommendations of Reference 2.

APPLICABLE SAFETY ANALYSES

The presence of a loose part in the RCS can be indicative of degraded reactor safety resulting from failure or weakening of a safety-related component. A loose part, whether it be from a failed or weakened component, or from an item inadvertently left in the primary system during construction, refueling, or maintenance, can contribute to component damage and material wear by frequent impacting with other parts in the system. Also, a loose part increases the potential for control-rod jamming and for accumulation of increased levels of radioactive crud in the primary system (Ref. 2).

The Loose Part Detection System provides the capability to detect loose parts in the RCS which could cause damage to some component in the RCS. Loose parts are not assumed to initiate any DBA, and the detection of a loose part is not required for mitigation of any DBA (Ref. 3).

(cont

BASES (continued)

APPLICABLE SAFETY ANALYSIS

Pipe and equipment supports, in general, are not directly considered in designing the accident sequences for theoretical hazard evaluations. Further, various Probabilistic Risk Assessment (PRA) studies have indicated that snubbers are not of prime importance in a risk significant sequence (Ref. 4 and 5). Therefore, the function of the snubbers is not essential in mitigating the consequences of a DBA or transient (Ref. 6).

TR

TR 3.7.3 requires that all snubbers utilized on safety related equipment shall be OPERABLE. Those snubbers that are utilized on non-safety related systems, shall be OPERABLE if a failure could have adverse effect on a safety related system. Individual snubbers may be removed from service for functional testing within the limits established herein without violating these requirements, although Required Actions and Completion Times still apply.

APPLICABILITY

The OPERABILITY of the snubbers is required in MODES 1, 2, 3, and 4. For MODES 5 and 6, the OPERABILITY is limited to the snubbers located on those systems which need to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1.1, A.1.2, and A.2

If one or more snubbers have been declared inoperable, the snubber(s) must be restored to OPERABLE status in 72 hours. Alternatively, the snubber(s) must be replaced in the 72 hours. In either case, an engineering evaluation per Table 3.7.3-5 must be performed during the 72 hours to:

a) Determine the cause of the failure

As a result of this evaluation, the need for testing other snubbers will be considered. The results from the testing will be used to consider expanded functional testing and cause examination with consideration of manufacturing and design deficiency. It should be noted that the testing must be independent and not combined with TSR 3.7.3.3.

(continued)

BASES

ACTIONS

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

b) Determine the impact on the supported component

This evaluation shall determine if the inoperable snubber has adversely affected the attached component.

The 72 hours is based on engineering experiences and is reasonable, considering the time it will take to identify the problem and take the proper corrective actions. This requirement is considered met for those snubbers rendered inoperable by removal for functional testing by the generic engineering evaluation included in Reference 9.

B.1

If Required Actions under Condition A are not met within the 72 hours, the supported system or component is immediately declared inoperable.

TECHNICAL SURVEILLANCE REQUIREMENTS

The TSRs are preceded by three Notes. Note 1 states that the snubber inservice inspection program shall be carried out in accordance with the requirements in Tables 3.7.3-1, 2 and 3. This represents an enhanced snubber inservice inspection program compared to the Inservice Inspection Program which stipulates inservice inspection in accordance with ASME section XI. The snubber inservice inspection program includes the requirements of Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions." ASME Section XI, 1989 Edition, Subsection IWF-5300(a) and (b) require that inservice examinations of snubbers, using the VT-3 visual examination method described in IWA-2213, and inservice tests of snubbers be performed in accordance with the first Addenda to ASME/ANSI OM-1987, Part 4. Note 2 requires repair or replacement of snubbers which fail inspection, and testing of repaired snubbers before installation. Note 3 indicates that a "snubber type," as used in this TR, is determined by the design and manufacturer, but not by size.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.3.1

TSR 3.7.3.1 comprises a visual inspection of the snubbers. A pre-fuel load visual inspection and functional test has been performed on each snubber using the acceptance criteria listed in this TSR. The baseline considers that the snubbers have experienced thermal cycling and normal operating service as a result of previous hot functional testing. The initial inservice inspection must be performed on the snubbers prior to completion of the first refueling outage. The frequency of subsequent surveillances depends on the number of snubbers found inoperable from each previous inspection as provided in Table 3.7.3-2 and the Inservice Inspection Program. The acceptance criteria and the remedial ACTIONS are listed in Table 3.7.3-1.

The visual inspections are designed to detect obvious indications of inoperability of the snubbers. Removal of insulation or direct contact with the snubbers is not required initially. However, suspected causes of inoperability are to be investigated and all snubbers of the same type and all snubbers subjected to the same failure mode are to be inspected more frequently.

The visual inspection frequency is based upon the number of unacceptable snubbers found during the previous inspection. Therefore, the required inspection intervals vary inversely with the number of inoperable snubbers found during an inspection. If a snubber fails the visual acceptance criteria, the snubber is declared unacceptable and cannot be declared OPERABLE via functional testing. However, if the cause of rejection is understood and remedied for that type of snubber and for any other type of snubbers, that may be generically susceptible, and OPERABILITY verified by testing, that snubber may be reclassified acceptable for the purpose of establishing the next surveillance interval.

Snubbers maybe categorized, according to accessibility, as noted in the Note to Table 3.7.3-2. The accessibility of each snubber is determined based on radiation level as well as other factors such as temperature, atmosphere, location, etc. The recommendations of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable" (Ref. 7) and Regulatory Guide 8.10, "Operation Philosophy for Maintaining Occupational Radiation Exposure as Low as Practicable" (Ref. 8), are considered in planning and implementing the visual inspection program.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.3.2

TSR 3.7.3.2 comprises the inspection of all snubbers attached to systems that have experienced unexpected, potentially damaging transients. The potential impact of the transients is assessed by reviewing operating data and by visually inspecting the associated systems. The review and the inspection must be performed within six months of the event. In addition to the inspection, the freedom-of-motion of the mechanical snubber(s) is verified in accordance with Table 3.7.3-3.

TSR 3.7.3.3

TSR 3.7.3.3 comprises the functional testing of hydraulic and mechanical snubbers. The testing for these snubbers have been separated into two sample plans. Sample plan A (10%) is used for the hydraulic steam generator snubbers based on the small population. Sample plan B (37) is used for mechanical snubbers based on the large population. The plans, when used in combination, are a conservative approach versus using only the sample plan B for the entire population.

Snubber functional testing is performed prior to completion of each refueling outage. This frequency is based on engineering experience and is reasonable for testing of a representative sample of snubbers. Credit may be taken toward meeting minimum outage testing requirements for mechanical snubbers functionally tested within the refueling cycle. Snubbers may be removed from service for functional testing in Modes 1 through 4 provided that the following administrative controls are implemented:

1. Required Actions and Completion Times must be met.
2. No more than one snubber may be removed from service at a time on any line and attached piping which is analyzed as a seismic subsystem. Multiple snubbers may be removed for testing simultaneously only if separated by two or more seismic anchors.
3. Snubbers on trained systems or portions of systems may be removed only on the train which is undergoing maintenance in that work week. Snubbers on non-trained systems or portions of systems may only be removed following a documented risk assessment. Snubbers may not be removed from service for testing on one train of a system while the other train has been declared inoperable for any reason.
4. Snubbers adjacent to equipment nozzles may not be removed for testing except in Modes 5 and 6. In determining the applicability of this limitation, engineering judgment must be used regarding the placement of the snubber relative to the nozzle, the routing of the affected piping, and any other supports available to protect equipment function.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.3.4

The TSR is preceded by three Notes which underline the need for considering service life of sub-components and to replace these sub-components before the end of the respective service lives. The replacement of sub-components must be documented and the documentation retained for further reference. TSR 3.7.3.4 address the monitoring of the service life of the snubbers. The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. The expected service life is established by the manufacturer and is based on operating experience with critical snubber parts such as seals and springs in a radiation environment. The every refueling outage Frequency is based on engineering experience and is reasonable for the verification service life.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III and XI.

2. Regulatory Guide 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports".

3. Regulatory Guide 1.130, "Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports".

4. "Zion Probabilistic Safety Study", Commonwealth Edison Company, September 1981.

5. "Millstone Unit 3 Probabilistic Safety Study," North-east Utilities Company, August 1983.

6. NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of The Commission's Interim Policy Statement Criteria to Standard Technical Specifications, Attachment to letter dated May, 1988 from T.E. Murley, NRC to W.S. Wilgus, Chairman The B&W Owners Group.

7. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable".

8. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposure as Low as Practicable."

9. SA/SE WBPLCE-97-028-0, RIMS T28970829803.

B 3.7 PLANT SYSTEMS

B 3.7.4 Sealed Source Contamination

BASES

BACKGROUND

A sealed source is any byproduct, source, or special nuclear material that is encased in a capsule designed to prevent leakage or escape of the material (Ref. 1). Sealed sources are classified into three groups according to their use (sources in use, not in use, and startup sources and fission detectors) and may contain beta, gamma, or alpha emitting material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on Reference 2. Those sources that are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring, excore fission detector assemblies or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

APPLICABLE SAFETY ANALYSES

The sealed source contamination requirement ensures that leakage from sealed sources will not exceed allowable intake values. This TR is important to the safety of plant personnel, however it is not required to mitigate the consequences of a DBA or transient (Ref. 3).

TR

TR 3.7.4 requires that the removable contamination shall be less than 0.005 microcuries for each sealed source containing the following radioactive material:

a. Greater than 100 microcuries of beta and/or gamma emitting material; or

b. Greater than 5 microcuries of alpha emitting material.

APPLICABILITY

Since the limits on the removable contamination for each sealed source containing radioactive material are not MODE dependent, this TR is applicable at all times.

(continue

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.

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 experienc 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)