ENCLOSURE 3

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WBN TECHNICAL REQUIREMENTS MANUAL

TECHNICAL REQUIREMENTS

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BASES

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LIST OF MISCELLANEOUS REPORTS AND PROGRAMS

LIST OF ACRONYMS

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

TECHNICAL REQUIREMENTS

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Revision 5	08-29-97
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Revision 8	09-22-97
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Revision 18	08-03-99

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	$\leq$ 0.5 second[1]
b. Low	$\leq$ 0.5 second[1]
3. Power Range, Neutron Flux	
a. High Positive Rate	N.A.
b. High Negative Rate	Deleted
4. Intermediate Range, Neutron Flux	N.A.
5. Source Range, Neutron Flux	N.A.
6. Overtemperature ΔT	$\leq$ 8 seconds[1]
7. Overpower ΔT	$\leq$ 8 seconds[1]
8. Pressurizer Pressure	
a. Low	$\leq$ 2 seconds
b. High	$\leq$ 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.

TR 3.7 PLANT SYSTEMS

TR 3.7.2 Flood Protection Plan

TR 3.7.2 The flood protection plan shall be ready for implementation to maintain the plant in a safe condition.

APPLICABILITY: When one or more of the following conditions exist:

a. Extreme flood-producing rainfall conditions in the east Tennessee watershed, or

b. Receipt of notification from River System Operations (RSO) that predicted flood levels based on rainfall on the ground or potential failure problems with one or more dams combined with critical headwater elevations and flood-producing rainfall may result in subsequent issuance of a Stage I flood warning.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Stage I flood warning issued.	A.1	Be in at least MODE 3.	6 hours
	<u>AND</u>		
	A.2	Initiate and complete the Stage I flood protection plan. Completion not required if flood warning is retracted by RSO.	10 hours
	<u>AND</u>		
	A.3	Establish a SDM ≥ 5% $\Delta k/k$ and T_{avg} ≤ 350°F.	10 hours
	<u>AND</u>		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4.1	Verify that communications between the TVA RSO and the Watts Bar Nuclear Plant have been established.	10 hours
		<u>OR</u>	
	A.4.2	Initiate and complete the Stage II flood protection procedure. Completion is not required if flood warning is retracted by RSO.	27 hours
B. Stage II flood warning issued.	B.1	Initiate and complete the Stage II flood protection plan. Completion is not required if flood warning is retracted by RSO.	17 hours <u>OR</u> Prior to flooding of the site.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.2.1	--------------------NOTE-------------------- Only required when one of the applicability criteria is met. -- Establish and maintain communications between Watts Bar Nuclear Plant and TVA RSO Group.	3 hours

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TR 3.7 PLANT SYSTEMS

TR 3.7.3 Snubbers

TR 3.7.3 All snubbers utilized on safety related systems shall be OPERABLE. For those snubbers utilized on non-safety related systems, each snubber shall be OPERABLE if a failure of that snubber or the failure of the non-safety related system would have an adverse effect on any safety related system.

APPLICABILITY: MODES 1, 2, 3, and 4.
MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more snubber(s) inoperable.	A.1.1 Restore snubber(s) to OPERABLE status.	72 hours
	OR	
	A.1.2 Replace snubber(s).	72 hours
	AND	
	A.2 Perform an engineering evaluation per Table 3.7.3-5 on the attached component.	72 hours

(continued)

TR 3.8 ELECTRICAL POWER SYSTEMS

TR 3.8.4 Submerged Component Circuit Protection

TR 3.8.4 The submerged component circuits associated with valves 1-FCV-74-1, 1-FCV-74-2, 1-FCV-74-8 and 1-FCV-74-9 shall be de-energized and the submerged components circuits associated with each component as shown in Table 3.8.4-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more submerged components circuits associated with valves 1-FCV-74-1, 1-FCV-74-2, 1-FCV-74-8, and 1-FCV-74-9 energized with RCS pressure ≥ 425 psig or one or more submerged components circuits inoperable for components listed in Table 3.8.4-1.	A.1 Restore the inoperable circuit to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.8.4.1	Verify that valves 1-FCV-74-1, 1-FCV-74-2, 1-FCV-74-8 and 1-FCV-74-9 are de-energized.	31 days
TSR 3.8.4.2	Verify that the components as shown in Table 3.8.4-1 are automatically de-energized on a simulated accident signal and that the components remain de-energized when the accident signal is reset.	18 months

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

* No adverse impact on power supplies if energized after accident signal reset.

BASES (continued)

TR

TR 3.3.6 requires the Loose-Part Detection System to be OPERABLE. This is necessary to ensure that sufficient capability is available to detect loose metallic parts in the RCS and avoid or mitigate damage to the RCS components. This requirement is provided in Reference 2.

APPLICABILITY

TR 3.3.6 is required to be met in MODES 1 and 2 as stated in Reference 2. These MODES of applicability are provided in Reference 2.

The Applicability has been modified by a Note stating that the provisions of TR 3.0.3 do not apply.

ACTIONS

A.1

If one or more required channels of the Loose-Part Detection System are inoperable for more than 30 days. the reactor need not be shutdown. but a report must be prepared and submitted to the NRC within the next 10 days in accordance with 10 CFR 50.4. This report is to outline the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status. This Condition. Required Action. and Completion Time are provided in Reference 2.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.6.1

Performance of a CHANNEL CHECK for the Loose-Part Detection System once every 24 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of even something more serious. CHANNEL CHECK will detect gross channel failure: thus. it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.6.1 (continued)

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal-processing equipment has drifted outside its limit.

The Surveillance and the Surveillance Frequency are provided in Reference 2.

TSR 3.3.6.2

A CHANNEL OPERATIONAL TEST is to be performed every 31 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the Loose-Part Detection System to detect impact signals which would indicate a loose part in the RCS. The Surveillance and the Surveillance Frequency are provided in Reference 2.

TSR 3.3.6.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The Surveillance Frequency of 18 months is based upon operating experience and is consistent with the typical industry refueling cycle. The Surveillance and the Surveillance Frequency are provided in Reference 2. Reference 1 describes the use of a computer-based analytical system to verify proper channel calibration. This is an acceptable option to using a mechanical impact device for sensors located in plant areas where plant personnel radiation exposure is considered by Plant Management to be excessive.

REFERENCES

1. Watts Bar FSAR, Section 7.6.7, "Loose Part Monitoring System (LPMS) System Description."

2. Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

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

BASES (continued)

REFERENCES

1. Watts Bar FSAR, Section 6.7.15, "Ice Condenser Instrumentation"
2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Inlet Door Position Monitoring System

BASES

BACKGROUND	Ninety-six limit switches monitor the position of the lower inlet doors. Two switches are mounted on the door frame for each door panel. The position and movement of the switches are such that the doors must be effectively sealed before the switches are actuated. A single annunciator window in the control room gives a common alarm signal when any door is open. Open/shut indication is also provided at the lower inlet door position display panel located in the Main Control Room. For door monitoring purposes, the ice condenser is divided into six zones, each containing four inlet door assemblies, or a total of eight door panels. The limit switches on the doors in any single zone are wired to a single light on the inlet door position display panel such that a closed light indicates that all the doors in that zone are shut and an open light indicates that one or more doors in that zone are open (Ref. 1). The display panel is considered the Inlet Door Position Monitoring System. Monitoring of inlet door position is necessary because the inlet doors form the barrier to air flow through the inlet ports of the ice condenser for normal unit operation. Failure of the Inlet Door Position Monitoring System requires an alternate OPERABLE monitoring system to be used to ensure that the ice condenser is not degraded.
APPLICABLE SAFETY ANALYSES	Proper operation of the inlet doors is necessary to mitigate the consequences of a loss of coolant accident or a main steam line break inside containment. The Inlet Door Position Monitoring System, however, is not required for proper operation of the inlet doors, nor is it considered OPERABLE as an initial condition for a DBA. Hence, the Inlet Door Position Monitoring System is not a consideration in the analyses of DBAs. Based on the PRA Summary Report in Reference 2, the Inlet Door Position Monitoring System has not been identified as a significant risk contributor.

(continued)

BASES (continued)

TR

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

APPLICABILITY

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

ACTIONS

A.1 and A.2

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

B.1

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

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

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

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.6.2.1

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

TSR 3.6.2.2

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

(continued)

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS	TSR 3.7.1.1 TSR 3.7.1.1 verifies that the pressures on the primary and the secondary sides in the steam generators are less than 200 psig. At temperatures below 70°F, the temperature margin to RT_{NDT} is diminished. Hence, the pressure must be checked every hour to ensure that the material toughness criteria are not violated. The 1-hour Frequency is based on engineering judgment and is consistent with industry practice.
REFERENCES	1. WCAP-13146, "Technical Basis for Determination of Secondary Side Pressure Test Temperatures in Sequoyah and Watts Bar Unit 1 and 2 Steam Generators." 2. 10 CFR 50, Appendix G, "Fracture Toughness Requirements." 3. ASME Boiler and Pressure Code, Section III. 4. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.7 PLANT SYSTEMS

B 3.7.2 Flood Protection Plan

BASES

BACKGROUND

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

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

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

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

(continued)

BASES

BACKGROUND (continued)	System, and relieving steam to the atmosphere through the steam generator power operated relief valves. Other essential plant cooling loads will be transferred from the Component Cooling Water System to the Essential Raw Cooling Water System (ERCW); the ERCW will also replace the Raw Cooling Water System to the ice condensers. The Radioactive Waste System will be secured by filling tanks below Design Bases Flood (DBF) level with enough water to prevent floatation; one exception is the waste gas decay tanks, which are sealed and anchored against floatation. Power and communication lines running beneath the DBF that are not required for submersed operation will be disconnected, and batteries below the DBF will be disconnected (Ref. 2).
APPLICABLE SAFETY ANALYSES	The flood protection plan specifies flood control measures to protect safety related equipment in the event that the maximum elevation for the ultimate heat sink or other body of water, as applicable, is exceeded. Because external flooding conditions present substantial warning time to achieve plant shutdown, this requirement is not a contributor to a dominant risk sequence (Ref. 3).
TR	TR 3.7.2 requires that the flood protection plan be ready for implementation to maintain the plant in a safe condition. This requirement ensures that facility protective actions will be taken and operation will be terminated in the event of flood conditions.
APPLICABILITY	The flood protection plan TR is applicable when one or more of the following conditions exist: a. Extreme flood producing rainfall conditions in the east Tennessee watershed, or b. Receipt of notification from TVA River System Operations (RSO) that predicted flood levels based on rainfall on the ground or potential failure problems with one or more dams combined with critical headwater elevations and flood-producing rainfall may result in subsequent issuance of a Stage I flood warning.

(continued)

BASES

ACTIONS

A.1. A.2. and A.3

If a Stage I flood warning is issued, several actions are required to be taken. The first requires the plant to be placed in MODE 3 in 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

Upon issuance of a Stage I flood warning, initiate and complete the Stage I flood protection plan, which involves preparatory steps. The required Completion Time for this Required Action is 10 hours. The plant is also required to be brought from full power operation to a safe shutdown. This is accomplished by Required Action A.3. This Required Action requires the establishment of a SHUTDOWN MARGIN of at least 5% $\Delta k/k$ and T_{avg} less than or equal to 350°F. The Completion Time of 10 hours is reasonable to accomplish the required SHUTDOWN MARGIN and T_{avg}.

A.4.1 and A.4.2

Once a Stage I flood warning has been issued, it is necessary to maintain communications between the TVA RSO Group and the Watts Bar Nuclear Plant. This is necessary because the TVA RSO Group provides the flood forecasting for the Watts Bar Nuclear Plant. The Completion Time of 10 hours corresponds to the time specified to initiate and complete the Stage I flood protection plan.

If communications between the TVA RSO Group and the Watts Bar Nuclear Plant have not been established within the required Completion Time, the Stage II flood protection procedure must be initiated and completed within 27 hours. The Completion Time of 27 hours corresponds to the minimum preflood preparation time. This is to ensure adequate warning time for safe plant shutdown.

B.1

If the Stage II flood warning has been issued, the Stage II flood protection plan must be initiated and completed within 17 hours or prior to flooding of the site. The Completion Time of 17 hours corresponds to the remaining hours of the 27 hour preflood preparation time after the Stage I flood warning consisting of 10 hours has expired, and is an adequate time period to complete Stage II preparations.

At any time it is determined that the potential for flooding at the site does not exist, the Stage I and Stage II flood protection plans are to be terminated immediately.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.2.1

This surveillance requires communications between Watts Bar Nuclear Plant and TVA RSO Group be established and maintained every 3 hours. A Note for this surveillance states that this is required only when one of the applicability criteria is met. This communications requirement exists because the TVA RSO Group provides the flood forecasting for Watts Bar Nuclear Plant. The 3 hour Frequency is adequate for early flood forecasting.

REFERENCES

1. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants."

2. Watts Bar FSAR, Section 2.4.14, "Flooding Protection Requirements."

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

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B 3.7 PLANT SYSTEMS

B 3.7.3 Snubbers

BASES

BACKGROUND

Component standard supports, are those metal supports which are designed to transmit loads from the pressure-retaining boundary of the component to the building structure. Although classified as component standard supports, snubbers require special consideration due to their unique function. Snubbers are either operated hydraulically or mechanically, depending on the nature of the support needed. They are designed to provide no transmission of force during normal plant operations, but function as a rigid support when subjected to dynamic transient loadings. Therefore, snubbers are chosen in lieu of rigid supports where restricting thermal grow during normal operation would induce excessive stresses in the piping nozzles or other equipment. The location and size of the snubbers are determined by stress analysis. Depending on the design classification of the particular piping, different combinations of load conditions are established. These conditions combine loading during normal operation, seismic loading and loading due to plant accidents/transients to four different loading sets. These loading sets are denominated: normal, upset, emergency, and faulted condition. The actual loading included in each of the four conditions, depends on the design classification of the piping. The calculated stresses in the piping and other equipment, for each of the four conditions, must be in conformance with established design limits.

Supports for pressure-retaining components are designed in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Division 1 (Ref. 1). The combination of loadings for each support, including the appropriate stress levels, meet the criteria of Regulatory Guide 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports" (Ref. 2), and Regulatory Guide 1.130, "Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports" (Ref. 3).

(continued)

BASES (continued)

ACTIONS

A.1

With the circuits for valves 1-FCV-74-1, 1-FCV-74-2, 1-FCV-74-8 and 1-FCV-74-9 energized with RCS pressure ≥ 425 psig or with one or more submerged components circuits (Table 3.8.4-1) inoperable, the associated submerged components could remain energized in the event of an accident. In order to prevent the adverse effects of a potential fault on an energized submerged component during an accident, it is necessary to restore the ability to automatically de-energize the component under accident conditions or to maintain a de-energized configuration for components not required to be functional. This can be done by restoring the inoperable circuit to OPERABLE status. The Completion Time of 7 days takes into consideration the low probability of an accident occurring which would cause the components to be submerged. It is a reasonable amount of time to complete the work necessary to restore the circuits to OPERABLE status.

B.1 and B.2

If the submerged component circuits cannot be restored to OPERABLE status within the 7 day Completion Time, it is necessary to place the plant in a Condition where the function of the circuits is not needed. This can be accomplished by first placing the plant in MODE 3 and then in MODE 5. The Completion Time of 6 hours to reach MODE 3 and 36 hours to reach MODE 5, are considered to be reasonable times for placing the plant into a condition where the TR is not applicable in a controlled manner.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.8.4.1

This surveillance requires verification that valves 1-FCV-74-1, 1-FCV-74-2, 1-FCV-74-8, and 1-FCV-74-9 are de-energized by removal of power at the 480V motor control centers. These valves are required to be shut in MODES 1,2,3, and 4, and are interlocked so that they cannot be opened until RCS Pressure is reduced to < 425 psig. The Frequency of 31 days is considered reasonable based on plant operating experience.

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.8.4.2

This surveillance requires verification that the components shown in Table 3.8.4-1 are automatically de-energized on a simulated accident signal. Since the function of OPERABLE submerged component circuits for the valves shown in the Table is to de-energize the components under accident conditions, verification that the valves are, in fact, de-energized on a simulated accident signal also constitutes verification that the submerged component circuits are OPERABLE. The 18 month Frequency corresponds to the availability of the components for testing during plant refueling.

REFERENCES

1. Watts Bar FSAR, Section 8.3.1.2.3, "Safety-Related Equipment in a LOCA Environment."

2. Watts Bar FSAR, Table 8.3-14, "Major Non-Safety Related Electrical Equipment That Could Become Submerged Following A LOCA."

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

4. WCAP-13470, "Watts Bar Unit 1 Technical Specifications Criteria Application Report," dated August, 1992.

5. Watts Bar Electrical Calculation WBNEEBMSTI080009.

ATTACHMENT

TRM CHANGED PAGES - SUPERSEDED

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.8.4.1	Verify that valves 1-FCV-74-1, 1-FCV-74-2, 1-FCV-74-8 and 1-FCV-74-9 are de-energized.	31 days
TSR 3.8.4.2	-------------------NOTE------------------ As noted in Table 3.8.4-1, some components are secured prior to resetting the accident signal. --- Verify that the components as shown in Table 3.8.4-1 are automatically de-energized on a simulated accident signal and that the components remain de-energized when the accident signal is reset.	18 months

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

* Are secured prior to resetting accident signal

** No adverse impact on power supplies if energized after accident signal reset.

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

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 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 REACTOR MOV BD 1A1-A

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

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

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

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 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 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 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 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 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

* Are secured prior to resetting accident signal

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

* Are secured prior to resetting accident signal

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.8.4.2

This surveillance requires verification that the components shown in Table 3.8.4-1 are automatically de-energized on a simulated accident signal. Since the function of OPERABLE submerged component circuits for the valves shown in the Table is to de-energize the components under accident conditions, verification that the valves are, in fact, de-energized on a simulated accident signal also constitutes verification that the submerged component circuits are OPERABLE. The NOTE clarifies that some Table 3.8.4-1 components require securing (e.g., hand switch manipulation, etc.), to prevent component energization due to plant process conditions which may exist concurrent with accident signal reset. These actions are contained in the applicable emergency procedures and addressed in Reference 6. The 18 month Frequency corresponds to the availability of the components for testing during plant refueling.

REFERENCES

1. Watts Bar FSAR, Section 8.3.1.2.3, "Safety-Related Equipment in a LOCA Environment."
2. Watts Bar FSAR, Table 8.3-14, "Major Non-Safety Related Electrical Equipment That Could Become Submerged Following A LOCA."
3. Watts Bar FSAR, Section 15.0, "Accident Analyses."
4. WCAP-13470, "Watts Bar Unit 1 Technical Specifications Criteria Application Report," dated August, 1992.
5. Watts Bar Electrical Calculation WBNEEBMSTI080009.
6. Watts Bar Nuclear Plant Safety Evaluation WBPLEE-99-039-2

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.8.4.2

This surveillance requires verification that the components shown in Table 3.8.4-1 are automatically de-energized on a simulated accident signal. Since the function of OPERABLE submerged component circuits for the valves shown in the Table is to de-energize the components under accident conditions. verification that the valves are. in fact. de-energized on a simulated accident signal also constitutes verification that the submerged component circuits are OPERABLE. The NOTE clarifies that some Table 3.8.4-1 components require securing (e.g.. hand switch manipulation. etc.). to prevent component energization due to plant process conditions which may exist concurrent with accident signal reset. These actions are contained in the applicable emergency procedures and addressed in Reference 6. The 18 month Frequency corresponds to the availability of the components for testing during plant refueling.

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5. Watts Bar Electrical Calculation WBNEEBMSTI080009.
6. Watts Bar Nuclear Plant Safety Evaluation WBPLEE-99-039-1

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.8.4.2

This surveillance requires verification that the components shown in Table 3.8.4-1 are automatically de-energized on a simulated accident signal. Since the function of OPERABLE submerged component circuits for the valves shown in the Table is to de-energize the components under accident conditions, verification that the valves are, in fact, de-energized on a simulated accident signal also constitutes verification that the submerged component circuits are OPERABLE. The NOTE clarifies that some Table 3.8.4-1 components require securing (e.g., hand switch manipulation, etc.), to prevent component energization due to plant process conditions which may exist concurrent with accident signal reset. These actions are contained in the applicable emergency procedures and addressed in Reference 6. The 18 month Frequency corresponds to the availability of the components for testing during plant refueling.

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3. Watts Bar FSAR, Section 15.0, "Accident Analyses."
4. WCAP-13470, "Watts Bar Unit 1 Technical Specifications Criteria Application Report," dated August, 1992.
5. Watts Bar Electrical Calculation WBNEEBMSTI080009.
6. Watts Bar Nuclear Plant Safety Evaluation WBPLEE-99-039-0