ENCLOSURE 3

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P PDR

ITS Section 1.0

ID 104

Subject: Deleted Section 1.0, Attachment 5, JFD 9 and replaced with TSTF 213 and revised NUREG Markup page 1.1-2 to remove JFD 9 annotation and replace with TSTF 213.

CTS

1.1 Definitions

TSTF 19, R1

CHANNEL CALIBRATION (continued)

sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated. 1.5.4

Supp 1

The CHANNEL CALIBRATION shall also include testing of safety related Reactor Protection System (RPS), Engineered Safety Feature Actuation System (ESFAS), and Emergency Feedwater Initiation and Control (EFIC) bypass functions for each channel affected by the bypass operation. TSTF 124

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter. 1.5.3

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display, and trip functions. 1.5.1 1.5.2 TSTF-205

Supp 1

of all devices in the channel required for channel OPERABILITY

The ESFAS CHANNEL FUNCTIONAL TEST shall also include testing of ESFAS safety related bypass functions for each channel affected by bypass operation. TSTF 124

Supp 2

CONTROL RODS

CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations. Doc A11 TSTF 213

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suppension of CORE 1.2.7 TSTF 205

Supp 1

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channell steps so that the entire channel is tested

Supp 1

(continued)

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1. The brackets are removed and the proper plant specific information or value is provided.

2. Not used.

3. Not used.

4. Not used.

5 The definition of AXIAL POWER SHAPING RODS (APSRs) has been modified to specify that these are control components with part-length absorbers. The ONS design provides control components with part-length absorbers. This specifically excludes the full length control components (regulating rods) when they are being used to control the axial power distribution of the reactor.

6 Not used.

7 Not used.

8 Not used.

2 9 Not used.

10 Not used.

11 The CLB regarding the half lives of nuclides included in the E-bar determination is retained. The definition is changed from including nuclides with half lives longer than 15 minutes to including nuclides with half lives longer than 30 minutes. The CLB definition is retained in order to maintain consistency in the approach to determining offsite doses for certain accident analyses.

12 Not used.

13 The definitions of EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME, ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME, and

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME were not incorporated. These terms and the referenced testing were not incorporated into ITS because they are not consistent with CTS. Response time testing of these systems, as required by specifications in NUREG 1430, is not required by CTS Specifications.

14 Not used.

15 Not used.

16 Not used.

17 Not used.

18 Not used.

19 The specific chapter reference in part "a." of the PHYSICS TESTS definition is deleted. ONS is not a Standard Review Plant. Consequently Physics Testing is described in more than one UFSAR Chapter. Removal of the reference to a specific chapter simply ensures that physics testing described in other Chapters of the UFSAR is encompassed by this definition.

20 ONS will maintain the RCS Pressure and Temperature Curves and Limits in the ITS and will not implement a PTLR at this time. Since a PTLR is not implemented, the definition serves no purpose and has been deleted.

21 Not used.

22 Not used.

23 The second reference provided in the NUREG definition for Dose Equivalent I-131 is deleted. UFSAR 15.14.7 uses the dose conversion factors specified in TID-14844 for the determination of Dose Equivalent I-131.

24 The subscript AVG is deleted to preclude in any potential confusion with Oconee instrumentation that measures reactor coolant temperature. Oconee has a T_{AVG} indication. However, this instrument cannot be used to measure reactor coolant temperature in the range specified in ITS Table 1.1-1.

25 The NUREG Definitions for NUCLEAR HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$ and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F_{\Delta H}^N$) are not adopted since the Specification, 3.2.5, "Power Peaking Factors" which uses these defined terms is not adopted.

26 An appropriate value of 250°F is selected for the MODE 4/MODE 3 transition temperature. There is no CTS equivalent value since the CTS does not include a similar MODE transition point. This temperature is appropriate since it is only slightly above the upper limit for using the Low Pressure Injection System in decay heat removal (DHR) alignment. This value permits use of MODE 4 as a transition MODE wherein required RCS flow may be maintained using either the RCS loops or the DHR loops.

27 The definition for L_a is not adopted in the ITS. The implementation of 10 CFR 50 Appendix J, Option B has resulted in modifications to the ISTS which capture the Leakage Rate Testing Program requirements in ITS paragraph 5.5.2. ITS section 5.5.16 provides a description of L_a which is consistent with the definition of L_a in ISTS Section 1.1, Definitions.

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Specification 3.0

(A1) <except as marked>

4 SURVEILLANCE REQUIREMENTS

4.0 SURVEILLANCE STANDARDS

Applicability

Applies to surveillance requirements which relate to tests, calibrations and inspections necessary to assure that the quality of structures, systems and components is maintained and that operation is within the safety limits and limiting conditions for operation.

Objective

To specify minimum acceptable surveillance requirements.

Specification

(M2)

~~4.0.1~~ Surveillance of structures, systems, components and parameters shall be as specified in the various subsections to this Technical Specification section, Section 4.0, except as permitted by Technical Specifications 4.0.2 and 4.0.3 below.

SR 3.0.1

~~4.0.2~~ Minimum surveillance frequencies, unless specified otherwise, may be adjusted as follows to facilitate test scheduling:

SR 3.0.2

Specified Frequency	Maximum Allowable Interval Between Surveillances
Five times per week	2 days
Two times per week	5 days
Weekly	10 days
Bi-Weekly	20 days
Monthly	45 days
Bi-Monthly	90 days
Quarterly	135 days
Semiannually	270 days
Annually	18 months
18 months	22 months, 15 days
Refueling Outage	22 months, 15 days

(M3)

Supp 2

Clarifying words in individual specifications such as "every," "at least," or "at least once every" are not intended to alter the frequencies defined by this specification.

~~4.0.3~~ If conditions exist such that surveillance of an item is not necessary to assure that operation is within the safety limits and limiting conditions for operation, surveillance need not be performed if such conditions continue for a length of time greater than the specified surveillance interval. Surveillance waived as a result of this specification shall be performed prior to returning to conditions for which the surveillance is necessary to assure that operation is within safety limits and limiting conditions for operation.

SR 3.0.1

SR 3.0.4

(A8)

4.0.4 Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

(SEE S.0)

Amendment No. 228(Unit 1)
Amendment No. 229(Unit 2)
Amendment No. 225(Unit 3)

Add SR 3.0.3 — (M2)

4 of 4

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Add SR 3.1.4.1 – M9

A1

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency	
SR 3.1.4.2	1. Control Rod Movement (1)	Movement of Each Rod	~~Monthly~~ 92 days – L5	
Supp 2	2. Pressurizer Safety Valves	Setpoint	18 months(4)	(SEE 3.4)
Supp 2	3. Main Steam Safety Valves	Setpoint	18 months(4)	(SEE 3.7)
	4. Refueling System Interlocks(5)	Functional	Prior to Refueling	(SEE 3.3)
	5. Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly	(SEE 3.7)
	6. Reactor Coolant System (2) Leakage	Evaluate	Daily	(SEE 3.4)
Supp 2	7. Emergency Condenser (6) Circulating Water System Test	Functional	18 months	(SEE 3.7)
	8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	
	9. Spent Fuel Cooling System	Functional	Prior to Refueling	
	10. High Pressure and Low (3) Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	(SEE 3.5)
Supp 2	11. Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	(SEE 3.7)
	12. MSLB Feedwater Isolation (7) Feature	Functional	18 months	
	13. Essential Siphon Vacuum (8) System Test	Functional	Quarterly	

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

TSC 95-03
Page 4 of 5

Specification 3.1.4

SR Applic (1) Applicable only when the reactor is critical. MODES 1 & 2 — M17

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. <SEE 3.4>

(3) Operating pumps excluded. <SEE 3.5>

<SEE 3.4 & 3.7>

Supp 2 | (4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

<SEE 3.3>

(5) Applicable only to the interlocks associated with the Reactor Building Purge System.

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature.

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit.

<SEE 3.7>

Specification 3.1.4

(A1) Except as Marked

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Trip Insertion Time Test

(L6) Add SR 3.1.4.3 Note

Applicability

Applies to the surveillance of the control rod trip insertion time.

Objective

To assure the control rod trip insertion time is within that used in the safety analyses.

Specification

SR 3.1.4.3

The control rod insertion time shall be measured at either full flow or no flow conditions as follows:

a. For all rods following each removal of the reactor vessel head, prior to criticality (M11)

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and (LA6)

c. For all rods at least once every 18 months. prior to criticality (M11) (Supp 2) (1)

The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66* seconds at reactor coolant full flow conditions or 1.40 seconds for no flow conditions. (M10)

For the APSRs it shall be demonstrated that loss of power will not cause rod movement. (LA2)

If the trip insertion time above is not met, the rod shall be declared inoperable.

* - For Unit 1 Cycle 15, Group 1, Rod 8 and Group 2, Rod 5 may be considered operable with an insertion time ≤ 3.00 sec provided: (A8)

1) the average insertion time for the remaining rods in Groups 1 and 2 is ≤ 1.50 sec, and

2) the core average negative reactivity insertion rate is within the assumptions of the safety analysis.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR Chapter 15.

A rod is considered inoperable if the trip insertion time is greater than the specified allowable time or the core average negative reactivity insertion rate is less than the assumptions of the safety analysis. (A2)

REFERENCES

(1) FSAR, Section 15
(2) Technical Specification 3.5.2

5 of 5

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

(A1) <except as marked>

TABLE 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	18 months	
b. Reactor Building Pressure (4 psig)	ES	MO	18 months	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	18 months	
20. Reactor Building Spray System Logic	NA	MO	NA	
21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	18 months	
22. Pressurizer Temperature	ES	NA	18 months	
23. Control Rod Absolute Position	ES (1)	NA	18 months (2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.

Supp 2

Supp 2

Supp 2 Page 2 of 2

<SEE 3.3>

SR 3.1.7.1

A2

A2

LA2

Specification 3.1.7

(A1) <except ● marked>

~~Table 4.1-1 (CONTINUED)~~

SR 3.1.7.1

~~Channel Description~~	~~Check~~	~~Test~~	~~Calibrate~~	Remarks
24. Control Rod Relative Position	ES (A12)	NA	18 months (2)	(1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel. (A12) (A12)
25. Core Flood Tanks				
a. Pressure	ES	NA	18 months	
b. Level	ES	NA	18 months	
26. Pressurizer Level	ES	NA	18 months	
27. Letdown Storage Tank Level	DA	NA	18 months	
28. Delete				
29. High and Low Pressure Injection Systems Flow Channels	NA	NA	18 months	<SEE 3.3>
30. Borated Water Storage Tank Level Indicator	WE	NA	18 months	
31. Boric Acid Mix Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
32. Concentrated Boric Acid Storage Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	

Supp 2

Page 2a of 2

Specification 3.1.7

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

ITS Section 3.1

ID 77

Subject: Revised ITS LCO 3.1.8 part c to specify the wide range instrumentation as providing the high startup rate CONTROL ROD withdrawal inhibit

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Safety Rod Position Limits";
LCO 3.2.1, "Regulating Rod Position Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoint set to ≤ 5% RTP;

2 c. Nuclear instrumentation wide range high startup rate CONTROL ROD withdrawal inhibit is OPERABLE;

d. SDM is within the limit specified in the COLR; and

e. RCS lowest loop average temperature is ≥ 520°F.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. THERMAL POWER not within limit.	A.1	Open control rod drive trip breakers.	Immediately
B. SDM not within limit.	B.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	B.2	Suspend PHYSICS TESTS exceptions.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Nuclear overpower trip setpoint is not within limit. OR Nuclear instrumentation wide range high startup rate CONTROL ROD withdrawal inhibit inoperable.	C.1 Suspend PHYSICS TESTS exceptions.	1 hour
D. RCS lowest loop average temperature not within limit.	D.1 Suspend PHYSICS TESTS exceptions.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify the RCS lowest loop average temperature is $\geq$ 520°F.	30 minutes

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

minimum RCS temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables.

PHYSICS TESTS Exceptions – MODE 2 satisfy Criteria 1, 2, and 3 of 10 CFR 50.36 (Ref. 6).

LCO

This LCO permits individual CONTROL RODS to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics.

This LCO also allows suspension of LCO 3.1.3, LCO 3.1.5, LCO 3.2.1, and LCO 3.4.2, provided:

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ 5% RTP;

2

c. Nuclear instrumentation wide range high startup rate CONTROL ROD withdrawal inhibit is OPERABLE;

d. SDM is maintained within the limit specified in the COLR; and

e. RCS lowest loop average temperature is ≥ 520°F.

The limits of LCO 3.2.2 and LCO 3.2.3 are not exempted by this specification because they do not apply in MODE 2. Inhibiting CONTROL ROD withdrawal, based on startup rate, also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

C.1

If the nuclear overpower trip setpoint is > 5% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

2

If the nuclear instrumentation wide range high startup rate CONTROL ROD withdrawal inhibit functions are inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

D.1

If the RCS lowest loop average temperature is not within limit, then 30 minutes is allowed for the operator to restore the RCS lowest loop average temperature to within limit or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

(continued)

proposed ONS ITS. This loss in flexibility is considered more restrictive and is consistent with the NUREG. Measuring control rod insertion times during flow conditions is more conservative and more appropriate since testing with reactor pumps operating simulates a reactor trip under actual conditions.

M11 CTS 4.7.1.c requires CONTROL ROD drop time testing following each refueling outage. ITS SR 3.1.4.3 Frequency is "Prior to reactor criticality after each removal of the reactor vessel head." Measuring CONTROL ROD drop times prior to reactor criticality after each removal of the reactor vessel head ensures the reactor internals and control rod drive mechanism will not interfere with CONTROL ROD motion or CONTROL ROD drop time and allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. This frequency is considered more restrictive since it explicitly requires completion of this testing prior to criticality and is consistent with the NUREG.

M12 CTS 3.1.9 provides low power physics testing restrictions. Since no specific actions are provided to address failure to meet these restrictions, CTS would require entry into LCO 3.0, which requires the unit be placed in Hot Shutdown (comparable to ITS MODE 3) within 12 hours. ITS 3.1.8 ACTION A, B, and C are added. ITS ACTION A requires the control rod drive trip breakers be opened immediately if the thermal power limit is exceeded. ITS 3.1.8 Action B requires boration be initiated within 15 minutes and physics tests exceptions be suspended within 1 hour if the SDM limit is exceeded. ITS 3.1.8 ACTION C requires physics tests exceptions be suspended within 1 hour if the nuclear overpower trip setpoint is not within limits or the nuclear
2 instrumentation wide range high startup rate control rod withdrawal inhibit is inoperable. The proposed actions are more restrictive since their Completion Times are shorter than the 12 hours of CTS LCO 3.0. The proposed ACTIONS are consistent with the NUREG

CTS 3.1.7 imposes an MTC limit above 95% RTP. ITS LCO 3.1.3 has an applicability in MODES 1 and 2 for MTC. The physics test exception for MTC provided by ITS 3.1.8 is necessary only due to the more restrictive requirement on MTC imposed by ITS LCO 3.1.3. The proposed change is consistent with the NUREG.

M13 CTS specify alignment limits for movable control assemblies (safety and regulating) but do no specify alignment limits for axial power shaping rods (APSRs). ITS 3.1.6 specifies that each APSR be within 6.5% of its group average height during MODES 1 and 2. Action A requires AXIAL POWER IMBALANCE to be determined within 2 hours when one APSR is inoperable or not aligned to within 6.5% of its group height and within 2 hours after each APSR movement. Action B requires the reactor to be in MODE 3 if the Required Action and associated Completion Time is not met. SR 3.1.6.1 requires verification that the LCO is met periodically (every 12 hours). The LCO, Required Actions and Surveillance

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4 ~~"CONTROL ROD Group Alignment Limits"~~;
LCO 3.1.5, "Safety Rod ~~Insertion~~ Limits";
LCO 3.1.6 ~~"AXIAL POWER SHAPING ROD (APSR) Alignment Limits"~~;
LCO 3.2.1, "Regulating Rod ~~Insertion~~ Limits" for the restricted operation region only; and
~~[~~LCO 3.4.2, "RCS Minimum Temperature for Criticality"~~]~~

may be suspended, provided:

a. THERMAL POWER is ≤ 5% RTP;

b. Reactor trip setpoints on the OPERABLE nuclear overpower channels are set to ≤ ~~[~~5%~~]~~ RTP;

c. Nuclear instrumentation ~~source range and intermediate~~ range high startup rate CONTROL ROD withdrawal inhibit ~~are~~ is OPERABLE; and

d. SDM is ≥ ~~[~~1.0~~]~~% Δk/k within the limit specified in the COLR; and

e. RCS lowest loop average temperature is ≥ 520°F

APPLICABILITY: ~~MODE 2~~ during PHYSICS TESTS initiated in MODE 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open control rod drive trip breakers.	Immediately

(continued)

CTS 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. SDM not within limit.	B.1 Initiate boration to restore SDM to within limit.	15 minutes
	AND	
	B.2 Suspend PHYSICS TESTS exceptions.	1 hour
C. Nuclear overpower trip setpoint is not within limit.	C.1 Suspend PHYSICS TESTS exceptions.	1 hour
OR		
Nuclear instrumentation ~~source~~ wide range and ~~intermediate~~ range high startup rate CONTROL ROD withdrawal inhibit inoperable.		

Doc M12

Doc M12

Supp 2

18

Supp.1

INSERT 3.1-22A — TSTF-(TBD)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.9.1 Verify THERMAL POWER is ~~≤ 5%~~ RTP.	~~1~~ hour (26)
SR 3.1.~~9~~8.2 (2) Verify nuclear overpower trip setpoint is ~~≤ 25%~~ 5 RTP. (16)	Once within 8 hours prior to performance of PHYSICS TESTS. (15)

Doc M19.

(continued)

INSERT 3.1-22B — TSTF-(TBD)

BASES

APPLICABLE SAFETY ANALYSES (continued)

temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables.

PHYSICS TESTS satisfy Criteria 1, 2, and 3 of ~~the NRC Policy Statement~~. (4)

Exception - MODE 2

10 CFR 50.36 (Ref. 6)

LCO

This LCO permits individual CONTROL RODS to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics.

This LCO also allows suspension of LCO 3.1.3, ~~LCO 3.1.4,~~ LCO 3.1.5, ~~LCO 3.1.6,~~ LCO 3.2.1, and LCO 3.4.2, provided: (1)

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ (26)% RTP; and

c. Nuclear instrumentation ~~source range and intermediate range~~ wide range high startup rate CONTROL ROD withdrawal inhibit are OPERABLE; and

d. SDM is maintained ≥ ~~[1.0]% Δk/k~~ within the limit provided/specified in the COLR

The limits of LCO 3.2 (2)/(3) and LCO 3.2 (3)/(4) do not apply in MODE 2. Inhibiting CONTROL ROD withdrawal, based on startup rate, also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power.

are not exempted by this specification because they (4)

(1) e. RCS lowest loop average temperature is ≥ 520°F.

Supp 2/ (26)

TSTF-009, Rev. 1

APPLICABILITY

This LCO is applicable ~~in MODE 2~~ when the reactor is either ~~not~~ critical ~~or when~~ THERMAL POWER ~~is~~ ≤ 5% RTP. This LCO is applicable for ~~initial criticality or~~ low power testing, as defined by Regulatory Guide 1.68 (Ref. 3) ~~in MODE 1.~~ (1)

with

during PHYSICS TESTS initiated in MODE 2

subcritical or

INSERT B 3.1-54A

(continued)

a test exception LCO is not required or allowed

BASES

ACTIONS

C.1 (continued)

Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

Supp 2

If the nuclear instrumentation (wide range) source and intermediate range high startup rate CONTROL ROD withdrawal inhibit functions are inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

(1)

INSERT B 3.1-56A

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Performing a CHANNEL FUNCTIONAL TEST on each nuclear instrumentation source and intermediate range high startup rate CONTROL ROD withdrawal inhibit and nuclear overpower channel, ensures that the instrumentation required to detect a deviation from THERMAL POWER or to detect a high startup rate is OPERABLE. Performing the test once within 24 hours, prior to initiating PHYSICS TESTS, ensures that the instrumentation is OPERABLE shortly before PHYSICS TESTS begin and allows the operator to correct any instrumentation problems.

(27)

SR 3.1.9.2

Verification that THERMAL POWER is ≤ 5% RTP ensures that an adequate margin is maintained between the THERMAL POWER level and the nuclear overpower trip setpoint. Hourly verification is adequate for the operator to determine any change in core conditions, such as xenon redistribution occurring after a THERMAL POWER reduction, that could cause THERMAL POWER to exceed the specified limit.

(1)

(1)

INSERT B 3.1-56B

(continued)

ITS Section 3.1

ID 98

Subject: Revised JFD 6 to clarify that NUREG SR 3.2.4.1 is required at all times, including when operating with a misaligned CONTROL ROD.

TECHNICAL SPECIFICATIONS

1 The brackets are removed and the proper plant specific information or value is provided.

2 Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number (including renumbering due to deletion or addition), reference, system description, or analysis description.

3 In SR 3.1.3.1, the phrase "within the upper limit specified in the COLR" is changed to "within the limits specified in the COLR." The term "upper" is not needed since the ITS does not include a lower MTC limit (refer to JFD 4 below). The COLR provides the appropriate limits at all power levels.

4 NUREG SR 3.1.3.2 is not included as a requirement in the ITS. CTS does not specify a lower limit on MTC. Although ONS has performed a verification of EOC MTC as a one time commitment, this verification is not performed each cycle. The change is consistent with the current licensing basis.

5 The NUREG value for nuclear overpower trip setpoint is replaced with the appropriate plant specific value.

6 NUREG-1430 Specification 3.1.4 Required Action A.2.5 and associated Completion Time is not included in the ONS ITS. This item is not a CTS requirement for this Condition; and therefore, is not being adopted. By not adopting these changes, requirements consistent with current license bases are being maintained. Removing the requirement to perform NUREG SR 3.2.5.1 while operating with a misaligned CONTROL ROD is appropriate since the CTS and ITS require the performance of NUREG SR 3.2.4.1, determination of QUADRANT POWER TILT, at all times when above 20% RTP, including when operating with a misaligned CONTROL. Performance of NUREG SR 3.2.4.1 ensures that any adverse trends in QPT, whether due to a misaligned CONTROL ROD or any other cause, are recognized within 2 hours. It further provides reasonable assurance that the potentially non-conservative CONTROL ROD position is not adversely affecting the power distribution within the core.

7 Not used.

8 The portion of NUREG-1430 SR 3.1.4.2 that requires moving each CONTROL ROD "≥ 3%" is replaced by "by an amount in any direction sufficient to demonstrate the absence of thermal binding." CTS 4.1-2, Item 1 does not specify how much to move the CONTROL RODS. The minimum distance a CONTROL ROD must be moved during testing is different between units. This is due to design differences between Units 1 & 2 and Unit 3. The CONTROL RODS on Units 1 & 2 are moved further to exercise the CRD fully through an area of thermal binding inherent in the design. This minimum

ITS Section 3.1

ID 99

Subject: Revised SR 3.1.4.2 to include a general acceptance criteria: "by moving an individual CONTROL ROD that is not fully inserted by an amount in any direction sufficient to demonstrate the absence of thermal binding." Added acceptance criteria to Bases.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual CONTROL ROD positions are within 6.5% of their group average height.	12 hours
SR 3.1.4.2	Verify CONTROL ROD freedom of movement (trippability) by moving each individual CONTROL ROD that is not fully inserted by an amount in any direction sufficient to demonstrate the absence of thermal binding.	92 days
SR 3.1.4.3	----------NOTE---------- With rod drop times determined with at least one but less than four reactor coolant pumps operating, operation may proceed provided operation is restricted to the pump combination operating during the rod drop time determination or a pump combination providing less total reactor coolant flow. ---------- Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is $\leq$ 1.66 seconds from power interruption at the CONTROL ROD drive breakers to $\frac{3}{4}$ insertion (25% withdrawn position).	Prior to reactor criticality after each removal of the reactor vessel head

BASES

ACTIONS

D.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL RODS are aligned within 6.5% of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other CONTROL ROD position information that is continuously available to the operator in the control room, so that during actual CONTROL ROD motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by an amount in any direction sufficient to demonstrate the absence of thermal binding will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is determined to be trippable and aligned, the CONTROL ROD(S) is considered to be OPERABLE. At any time, if a CONTROL ROD(S) is immovable, a determination of the trippability (OPERABILITY) of the CONTROL ROD(S) must be made, and appropriate action taken.

2
2
2

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual CONTROL ROD positions are within [6.5]% of their group average height. (1)	4 hours when the asymmetric CONTROL ROD alarm is inoperable AND 12 hours when the asymmetric CONTROL ROD alarm is OPERABLE
SR 3.1.4.2	Verify CONTROL ROD freedom of movement (trippability) by moving each individual CONTROL ROD that is not fully inserted by an amount ≥ 3% in any direction. (8)	92 days
SR 3.1.4.3	----------NOTE---------- With rod drop times determined with less than four reactor coolant pumps operating, operation may proceed provided operation is restricted to the pump combination operating during the rod drop time determination sufficient to demonstrate the absence of thermal binding or a pump combination providing less total reactor coolant flow (22) ---------- Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is ≤ [1.66] seconds from power interruption at the CONTROL ROD drive breakers to ¾ insertion (25% withdrawn position) (1) with T_{avg} ≥ 525°F. (25)	at least one but Prior to reactor criticality after each removal of the reactor vessel head

Supp. 1

Supp. 2

Supp 2

Doc M9

TSTF 110, R2

T 4.1-2 Item 1

Doc L6

4.7.1

TECHNICAL SPECIFICATIONS

1 The brackets are removed and the proper plant specific information or value is provided.

2 Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number (including renumbering due to deletion or addition), reference, system description, or analysis description.

3 In SR 3.1.3.1, the phrase "within the upper limit specified in the COLR" is changed to "within the limits specified in the COLR." The term "upper" is not needed since the ITS does not include a lower MTC limit (refer to JFD 4 below). The COLR provides the appropriate limits at all power levels.

4 NUREG SR 3.1.3.2 is not included as a requirement in the ITS. CTS does not specify a lower limit on MTC. Although ONS has performed a verification of EOC MTC as a one time commitment, this verification is not performed each cycle. The change is consistent with the current licensing basis.

5 The NUREG value for nuclear overpower trip setpoint is replaced with the appropriate plant specific value.

6 NUREG-1430 Specification 3.1.4 Required Action A.2.5 and associated Completion Time is not included in the ONS ITS. This item is not a CTS requirement for this Condition; and therefore, is not being adopted. By not adopting these changes, requirements consistent with current license bases are being maintained. Removing the requirement to perform NUREG SR 3.2.5.1 while operating with a misaligned CONTROL ROD is appropriate since the CTS and ITS require the performance of NUREG SR 3.2.4.1,
2 determination of QUADRANT POWER TILT, at all times when above 20% RTP,
2 including when operating with a misaligned CONTROL. Performance of NUREG SR 3.2.4.1 ensures that any adverse trends in QPT, whether due to a misaligned CONTROL ROD or any other cause, are recognized within 2 hours. It further provides reasonable assurance that the potentially non-conservative CONTROL ROD position is not adversely affecting the power distribution within the core.

7 Not used.

8 The portion of NUREG-1430 SR 3.1.4.2 that requires moving each CONTROL
2 ROD "≥ 3%" is replaced by "by an amount in any direction sufficient to
2 demonstrate the absence of thermal binding." CTS 4.1-2, Item 1 does not specify how much to move the CONTROL RODS. The minimum distance a CONTROL ROD must be moved during testing is different between units. This is due to design differences between Units 1 & 2 and Unit 3. The CONTROL RODS on Units 1 & 2 are moved further to exercise the CRD fully through an area of thermal binding inherent in the design. This minimum

BASES

SURVEILLANCE REQUIREMENTS (continued)

Supp. 2

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by an amount in any direction sufficient to demonstrate the absence of thermal binding will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is determined to be trippable and aligned, the CONTROL ROD(S) is considered to be OPERABLE. At any time, if a CONTROL ROD(S) is immovable, a determination of the trippability (OPERABILITY) of the CONTROL ROD(S) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop time allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. The rod drop time given in the safety analysis is 1.66 seconds to 3/4 insertion, using the identical rod drop curve gives a value of 1.66 seconds to 3/4 insertion. The latter value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at 3/4 insertion to give an indication of the rod drop time and rod location. Measuring rod drop times, prior to reactor criticality after reactor vessel head removal and after CONTROL ROD drive system maintenance or modification, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or rod drop time. This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

(continued)

ITS Section 3.1

ID 105

Subject: Revised Section 3.1, Attach 3, DOC R1 to indicate document relocated to UFSAR Chapter 16.

RELOCATED SPECIFICATIONS

R1 CTS 3.5.2.2.b.6 and 4.7.2, which pertain to control rod program
2 verification, are relocated to UFSAR Chapter 16 because they have been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Station Unit 1, 2, and 3 Technical Specifications."

ITS Section 3.1

ID 119

Subject: Annotated TSTF-220 in the spot provided (blank bubble) on NUREG page 3.1-12, B 3.1-35, B 3.1-36.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

CTS

LCO 3.1.6 Each APSR shall be OPERABLE and aligned within ~~[6.5]~~ 4 % of its group average height. (1) DOC M13

Supp. 1 | APPLICABILITY: MODES 1 and 2 ~~when the APSRs are not fully withdrawn~~. TSTF-159, R1 DOC M13

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One APSR inoperable, not aligned within its limits, or both.	A.1 Align the APSR group to within [6.5]% of the inoperable or misaligned rod, while maintaining the APSR insertion limits in the COLR.	2 hours DOC M13
Supp. 1	Perform SR 3.2.2.1	Restore APSR alignment. TSTF 143
Supp 2 |	TSTF-220	
	AND	AND
	A.2 Prevent movement of the APSR group, while the rod remains inoperable or misaligned.	2 hours after each APSR movement
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	~~6~~ 12 – 27 hours DOC M13

(4) <Except as marked>

BASES

LCO (continued)

value is established based on the distance between reed switches, with additional allowances for uncertainty in the equipment used to determine this value. absolute position indicator amplifiers, group maximum or minimum synthesizer, and asymmetric alarm or fault detector outputs. The position of an inoperable ~~rod~~ is not included in the calculation of the ~~rod~~ group's average position. (14)

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

APPLICABILITY

The requirements on APSR OPERABILITY ~~and alignment~~ are applicable in MODES 1 and 2, when the APSRs are not fully ~~withdrawn~~ because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY and alignment of ~~rods~~ have the potential to affect the safety of the ~~plant~~. OPERABILITY and alignment of the APSRs are not required when they are fully withdrawn because they do not influence core power peaking. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down ~~and not producing fission power~~, and excessive local LHRs cannot occur from APSR misalignment.

(1) (1) (1) (

BASES

ACTIONS

A.1 ~~and A.2~~ (continued)

Required Action A.1 ~~is~~ only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

The reactor may continue in operation with the APSR misaligned if further movement of the APSR group is prohibited, so ~~that the misalignment~~ does not increase and cause the limits on AXIAL POWER IMBALANCE to be exceeded. The required Completion Time of up to 2 hours will not cause significant xenon redistribution to occur.

B.1

The ~~plant~~ must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the ~~plant~~ must be brought to at least MODE 3 within ~~6~~ hours. The Completion Time of ~~6~~ hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging ~~plant~~ systems. In MODE 3, APSR ~~group~~ alignment limits are not required because the reactor is not generating THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within [6.5]% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. If the asymmetric CONTROL ROD alarm is inoperable, a 4 hour Frequency is reasonable to prevent large deviations in APSR alignment from occurring without detection. In addition, APSR position is continuously available to the operator in the control room so that during actual ~~rod~~ motion, deviations can immediately be detected.

(continued)

ITS Section 3.1

ID 141

Subject: Revised Bases page B 3.1-36, last line to delete "or transferred" At ONS, there is no such indication.

BASES

BACKGROUND (continued)

Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD and APSR assembly (CRA) leadscrew extension comes near. As the leadscrew and CONTROL ROD or APSR move, the switches operate sequentially, producing an analog voltage proportional to position. Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications, and absolute position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven indicators are called zone reference indicators. The relative position indicator transducer is a potentiometer, driven by a pulse stepping motor that produces a signal proportional to CONTROL ROD or APSR position, based on the electrical pulse steps that drive the CRDM.

Two absolute position indicator channel designs may be used in the unit: type A absolute position indicators and type A-R4C absolute position indicators. The type A absolute position indicator transducer is a voltage divider circuit made up of 48 resistors of equal value connected in series. One end of 48 reed switches is connected at a junction between each of the resistors, so that as the magnet mounted on the leadscrew moves, either one or two reed switches are closed in the vicinity of the magnet. The type A-R4C (redundant four channel) absolute position indicator transducer has two parallel sets of voltage divider circuits made up of 36 resistors each, connected in series (channels A and B). One end of 36 reed switches is connected at a junction between each of the resistors of the two parallel circuits. The reed switches making up each circuit are offset, such that the switches for channel A are staggered with the switches for channel B. The type A-R4C is designed such that either two or three reed switches are closed in the vicinity of the magnet. By its design, the type A-R4C absolute position indicator provides redundancy, with the two three sequence of pickup and drop out of reed switches to enable a continuity of position signal when a single reed switch fails to close.

CONTROL ROD and APSR position indicating readout devices located in the control room consist of single rod position meters on a position indication panel. A selector switch permits either relative or absolute position indication to be displayed. Indicator lights are provided on the position indication panel to indicate when each CONTROL ROD or APSR is fully withdrawn, fully inserted, or enabled,

2

(continued)

(4) <Except as marked>

BASES

Supp 2

BACKGROUND (continued)

enabled, or transferred and whether a ~~CRA~~ Alternate rod position asymmetry alarm condition is present. Indicators ~~on the console~~ show full insertion, full withdrawal, and enabled ~~for motion~~ for each CONTROL ROD group. Identical ~~instrumentation and devices~~ exist for the APSR group and APSR under control. The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD position. Therefore, the potential for operation in violation of design peaking factors or SDM is increased. (1)

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. Regulating rod, safety ~~rod~~, and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod ~~Insertion~~ Limits"; LCO 3.2.1, "Regulating Rod ~~Insertion~~ Limits"; and ~~LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"~~). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"). CONTROL ROD and APSR positions are continuously monitored to provide operators with information that ensures the ~~plant~~ unit is operating within the bounds of the accident analysis assumptions. (4) The CONTROL ROD and APSR position indicator channels satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). The CONTROL ROD position indicators monitor CONTROL ROD position, which is an accident initial condition.

CONTROL ROD (Position) (1)

CONTROL ROD and APSR

LCO

LCO 3.1.7 specifies that one absolute position indicator channel and one relative position indicator channel be OPERABLE for each CONTROL ROD and APSR. (1)

The agreement between the relative position indicator channel and the absolute position indicator channel, within the limit given in the COLR, indicates that relative position indicators are adequately calibrated and can be

(continued)

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
33. Containment Temperature	NA	NA	18 months	
34. Incore Neutron Detectors	MO(1)	NA	NA	(1) Check functioning, including functioning of computer readout or recorder readout.
35. Emergency Plant Radiation Instruments	MO(1)	NA	18 months	(1) Battery check.
36. Environmental Monitors	MO(1)	NA	18 months	(1) Check functioning.
37. Reactor Manual Trip	NA	PS	NA	
38. Reactor Building Emergency Sump Level	NA	NA	18 months	
39. Steam Generator Water Level	WE	NA	18 months	
40. Turbine Overspeed Trip	NA	NA	18 months	
41. Engineered Safeguards Channel 1 HP Injection & Reactor Building Isolation Manual Trip	NA	18 months	NA	Includes Reactor Building isolation of non-essential systems only
42. Engineered Safeguards Channel 2 HP Injection & Reactor Building Isolation Manual Trip	NA	18 months	NA	Includes Reactor Building isolation of non-essential systems only
43. Engineered Safeguards Channel 3 LP Injection Manual Trip	NA	18 months	NA	

ITS Section 3.2

ID 78

Subject: Revised Spec 3.2.1 to consolidate ACTIONS B and C as recommended by NRC Reviewer.

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Regulating Rod Position Limits

LCO 3.2.1 Regulating rod groups shall be within the physical position, sequence, and overlap limits specified in the COLR.

----------NOTE----------
Not required for any regulating rod positioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Regulating rod groups sequence or overlap requirements not met.	A.1 Restore regulating rod groups to within limits.	2 hours

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Regulating rod groups positioned in restricted or unacceptable region.	B.1	--------NOTE-------- Not applicable to regulating rod groups positioned in the restricted region. ---------- Initiate boration to restore SDM to within the limits specified in the COLR.	15 minutes
	AND		
	B.2.1	Restore regulating rod groups to within acceptable region.	2 hours
	OR		
	B.2.2	Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group position limits.	2 hours
C. Required Action and associated Completion Time of Condition A or B or not met.	C.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	12 hours
SR 3.2.1.2	Verify regulating rod groups meet the position limits as specified in the COLR.	12 hours
SR 3.2.1.3	Verify SDM to be within the limit as specified in the COLR.	Within 4 hours prior to achieving criticality

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.2 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
2 A. AXIAL POWER IMBALANCE not within limits.	A.1 Restore AXIAL POWER IMBALANCE to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 40% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.3 QUADRANT POWER TILT (QPT)

LCO 3.2.3 QPT shall be maintained less than or equal to the steady state limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPT greater than the steady state limit and less than or equal to the transient limit.	A.1	Reduce THERMAL POWER ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	2 hours
	AND		
	A.2	Reduce nuclear overpower trip setpoints, based on flux and flux/flow imbalance, ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	10 hours
	AND		
	A.3	Restore QPT to less than or equal to the steady state limit.	24 hours from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. QPT greater than the transient limit and less than or equal to the maximum limit due to misalignment of a CONTROL ROD or an APSR.	B.1	Reduce THERMAL POWER ≥ 2% RTP from ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	30 minutes
	AND		
	B.2	Restore QPT to less than or equal to the transient limit.	2 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Reduce THERMAL POWER to < 60% of the ALLOWABLE THERMAL POWER.	2 hours
	AND		
	C.2	Reduce nuclear overpower trip setpoints, based on flux and flux/flow imbalance, to ≤ 65.5% of the ALLOWABLE THERMAL POWER.	10 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. QPT greater than the transient limit and less than or equal to the maximum limit due to causes other than the misalignment of either CONTROL ROD or APSR.	D.1 Reduce THERMAL POWER to < 60% of the ALLOWABLE THERMAL POWER. AND D.2 Reduce nuclear overpower trip setpoints, based on flux and flux/flow imbalance, to ≤ 65.5% of the ALLOWABLE THERMAL POWER.	2 hours 10 hours
E. Required Action and associated Completion Time for Condition C or D not met.	E.1 Reduce THERMAL POWER to ≤ 20% RTP.	4 hours
F. QPT greater than the maximum limit.	F.1 Reduce THERMAL POWER to ≤ 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify QPT is within limits as specified in the COLR.	7 days AND When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at ≥ 95% RTP

BASES

ACTIONS (continued)

A.1

Operation with the regulating rod groups out of sequence or with the group overlap limits exceeded may represent a condition beyond the assumptions used in the safety analyses. The design calculations assume no deviation in nominal overlap between regulating rod groups. However, small deviations in group overlap, as allowed by the COLR, may occur and would not cause significant differences in core reactivity, in power distribution, or rod worth, relative to the design calculations. Group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a predetermined order. The Completion Time of 2 hours is intended to restrict operation in this condition because of the potential severity associated with gross violations of group sequence or overlap requirements. The 2 hour Completion Time is based on operating experience which supports the restoration time without unnecessarily challenging unit operation and the low probability of an event occurring simultaneously with the limit out of specification.

2 B.1, B.2.1 and B.2.2

Operation in the unacceptable region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods
2 may be positioned too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit.
2 Therefore, Required Action B.1 requires the RCS boron concentration be increased to restore the regulating rod position to a value that preserves the SDM and ejected rod worth limits. The RCS boration must occur as described in
2 Section B 3.1.1. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted region, which restores the minimum SDM and reduces the potential ejected
2 rod worth to within its limit. Required Action B.1 is

(continued)

BASES

ACTIONS B.1, B.2.1 and B.2.2 (continued)

2 modified by a Note indicating that it is not applicable to
2 regulating rod groups positioned in the restricted region.

Indefinite operation with the regulating rods inserted in the restricted region or unacceptable region is not prudent. Reactivity limits may not be met and the abnormal regulating rod position or group configuration may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long
2 term fuel depletion pattern. Therefore, Required Action
2 B.2.1 requires restoration of regulating rod groups to within their limits within 2 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group positions or configurations, thereby limiting the potential for an adverse xenon redistribution.

The regulating rods can also be restored within the acceptable position limits by reducing the THERMAL POWER to a value allowed by the regulating rod position limits in the
2 COLR as allowed by Required Action B.2.2. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems.

2 C.1

If the Required Action and associated Completion Time of
2 Condition A or B are not met, then the reactor must be placed in MODE 3, a MODE in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 12 hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging unit systems.

(continued)

(A1) <except as marked>

3.2.
~~3.5.2~~ ~~Control Rod Group and~~ Power Distribution Limits

3.2.1 App

Applicability

This specification applies to power distribution and operation of control rods during power operation. (A9) (M1)

MODES 1 & 2

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

~~Specification~~

Supp 2 RA B.1

~~3.5.2.1~~ Shutdown Margin

a. The available shutdown margin shall be greater than 1% Δk/k with the highest worth control rod fully withdrawn.

With Regulating Rods inserted in unacceptable region

b. If the shutdown margin is less than 1% Δk/k, then within ~~1 hour~~ initiate and continue boration until the required shutdown margin is restored. The requirements of Specification 3.5.2.5.c shall be met. (A4) 15 minutes (M7)

RA B.2.1

3.5.2.2 Movable Control Assemblies

a. All control (safety and regulating) rods shall be operable and positioned within nine (9) inches of their group average height.

b. A control rod shall be declared inoperable if any of the following conditions exist for that rod:

1. The control rod cannot be moved due to excessive friction or mechanical interference, or cannot perform its intended trip function.

2. The control rod cannot be located by either absolute or relative position indication or by in or out limit lights.

3. The control rod is misaligned with its group average by more than nine (9) inches.

4. The control rod does not meet the exercise requirements of Specification 4.1.

5. The control rod does not meet the rod trip insertion times of Specification 4.7.1.

6. The control rod does not meet the rod program verification of Specification 4.7.2.

(SEE 3.1)

A 66/66/63
12/15/78

(A1) <except as marked>

<SEE 3.2.3>

3. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

e. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit provided in the Core Operating Limits Report, due to causes other than simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 2 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

f. If the maximum positive quadrant power tilt exceeds the Maximum Limit provided in the Core Operating Limits Report, the reactor shall be shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.

g. Quadrant power tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% full power.

~~3.5.2.5 Control Rod Positions~~

<SEE 3.1>

a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

<SEE 3.1>

LCO 3.2.1

b. Except for physics tests operating rod group overlap shall be ~~25%~~ ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours or the reactor shall be placed in a hot shutdown condition within an additional ~~12~~ hours.

(LA1)

Supp 2

Req. Act. A.1
Req. Act. C.1

MODE 3

within the limits specified in the COLR

(A1) <except as marked>

LCO 3.2.1 NOTE — SEE 3.1 — A8

LCO 3.2.1 Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits shall be maintained within acceptable operating limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT for the particular number of operating reactor coolant pumps (4,3). — A8, A6

Supp 2 | RA B.2.1 & B.2.2 | RA B.1

If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times. — A13, A11

3.5.2.6 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the acceptable operating limits for reactor power imbalance provided in the CORE OPERATING LIMITS REPORT.

SEE 3.1 — SEE 3.2.2

If the imbalance is not within the acceptable envelope, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate. — R1

Add SR 3.2.1.1 — M4

Add SR 3.2.1.2

Supp 2 | Add ACTION C — A5

Add SR 3.2.1.3 — M6

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with NUREG-1430, Revision 1. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430, Revision 1. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1430, Revision 1. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A2 Current technical specification (CTS) Bases are completely replaced by revised bases that reflect the format and applicable content of proposed ITS Section 3.2. The revised Bases are shown in the proposed ONS ITS Bases for Section 3.2.

A3 CTS 3.5.2.6 requires power be reduced until AXIAL POWER IMBALANCE limits are met if they are not restored in two hours. ITS requires the power be reduced to $\leq$ 40% RTP. The CTS actions are consistent with ITS 3.2.2 Action B with the exception that a final specific power level is not explicitly established in the CTS. The final power level is implicitly established by the Applicability criteria specified in CTS 3.5.2.6 in that the Specification applies during power operation above 40% RTP. Based on the current Applicability of CTS 3.5.2.6 and the requirements of CTS 3.0, the maximum required power reduction consists of placing the unit in a MODE in which the Specification no longer applies. In adopting this specified power level in CTS 3.5.2.6, the Required Actions are made explicit. This change constitutes an administrative change intended to provide clarification and explicit guidance in conformance with the format and presentation of the NUREG-1430. No technical or intent change is associated with this editorial specification of an explicit power level. The proposed change is consistent with the NUREG.

A4 CTS 3.1.3.5 establishes the LCO requirements for safety rod and regulating rod group positions and requires that the position satisfy CTS 3.5.2.1. CTS 3.5.2.1 requires the available SHUTDOWN MARGIN be greater than 1% $\Delta k/k$ with the highest worth CONTROL ROD fully withdrawn during power operation. In addition, CTS 3.5.2.1 establishes the Required Action if this SHUTDOWN MARGIN requirement is not satisfied (i.e., within 1 hour initiate and continue boration until the required SDM is met). CTS 3.1.11 requires available shutdown margin during all system conditions except refueling be > 1% $\Delta k/k$ with the highest worth CONTROL ROD fully withdrawn during power operation. If CTS 3.1.11 is

not met entry into CTS 3.0 is required. Therefore the more limiting actions would be to initiate a shutdown per CTS 3.0.

CTS 3.5.2.1 modifies the allowable position limits of CTS 3.1.3.5 for the safety and regulating rod groups based on the ability to satisfy an imposed SHUTDOWN MARGIN requirement defined within CTS 3.5.2.1. With respect to the safety rod groups, this limitation requires that SDM be evaluated any time a safety rod is less than fully withdrawn during an approach to criticality, or for that matter, with the reactor critical (comparable to ITS MODES 1 and 2)(Ref. CTS 3.1.3.5). The requirement for the safety rod groups are retained in ITS 3.1.5 and are discussed in the Discussion of Changes (DOCs) for Section 3.1. With respect to the regulating rod groups, this SDM requirement is maintained in the ITS through: 1) the basis for derivation of the regulating rod group position limits given in the COLR, and 2) the Required Actions for
2 initiation of boration to restore SDM within 15 minutes (LCO 3.2.1
2 Required Action B.1) and the Required Actions to restore regulating rod
2 groups to within the acceptable region (LCO 3.2.1 Required Action B.2.1) or reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group's position limits (LCO 3.2.1
2 Required Action B.2.2). This change represents a change in format only. Requirements consistent with the referenced portion of the CTS are maintained by the adoption of this ITS requirement. No additional requirements are added and no existing requirements are removed by this change. The proposed change is consistent with the NUREG with the exception to the modifications described in JFDs for the NUREG Specification 3.2.1.

A5 The CTS currently does not have a shutdown requirement specific to CTS 3.5.2.5.c when the requirements of 3.5.2.5.c (i.e., attaining an acceptable control rod position) are not met. Therefore, CTS 3.0
2 requirements apply. ITS ACTION C is added to provide an explicit requirement to be in MODE 3 within 12 hours of failure to restore sequence or position limits. This is equivalent to the shutdown time in CTS 3.0. Therefore, the proposed change is considered administrative and is consistent with the NUREG.

A6 CTS 3.5.2.5.c states that the regulating control rod position limits shall be maintained within acceptable operating limits for regulating control rod position provided in the COLR for the particular number of operating reactor coolant pumps (4,3). These limits are already in the COLR based on the number of operating reactor coolant pumps (RCP). Therefore, specific reference to the number of operating RCPs provides no requirement and has been deleted. Therefore, the proposed change is considered administrative and is consistent with the NUREG.

A7 CTS 3.5.2.4.b.2 for Quadrant Power Tilt (QPT) differentiates action to be taken when the maximum QPT exceeds the steady state limit but is less than or equal to the transient limit provided in the COLR based on reducing from 100% power or reducing from the allowable thermal power for the reactor coolant pump combination. Comparable ITS LCO 3.2.3

Required Action A.1 is based on reducing thermal power from the allowable thermal power. Since the CTS starting points are equivalent and the actions are the same (i.e, reduce thermal power by 2% for each 1% of excess tilt), this differentiation serves no purpose and has been eliminated. As such, the proposed change is considered administrative and is consistent with the NUREG.

A8 The first sentence of CTS 3.5.2.5.c implies that there are position limits for the axial power shaping rods (APSRs). However, there are no CTS position limits for APSRs or actions required if APSR position limits are not met. The APSR position limits were eliminated in a Technical Specification amendment associated with Oconee's change to the use of only gray (Inconel) APSRs, however, the amendment should have but did not remove the specific reference to APSRs position limits. Conversion to the gray APSRs eliminated the need to require APSR position limits. As such deletion of reference to APSR position limits is considered administrative.

A9 CTS 3.1.3.5 Applicability is "prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality." CTS 3.5.2 Applicability for rod position limits is during power operation. ITS 3.2.1 Applicability is MODES 1and 2. CTS 3.1.3.5 specifies a sequence of reactivity additions "during the approach to criticality," including deboration after withdrawal of safety rod groups. The CTS bases clarifies that this is done to ensure shutdown capability during startup. In this context, "approach to criticality" is the same as "startup." Since CTS 3.1.3.5 and 3.5.2 require compliance with regulating rod position limits while in startup reactor operating conditions (comparable to ITS MODE 2) as well as during power operations (comparable to ITS MODE 1), the adoption of the Applicability of ITS 3.2.1 maintains requirements consistent with the Applicability of CTS 3.1.3.5 and CTS 3.5.2. As such, the proposed change is considered administrative and is consistent with the NUREG.

A10 CTS 3.5.2.4.f directs that the Unit be shut down in 4 hours (reactor subcritical) if the QUADRANT POWER TILT (QPT) is in excess of the maximum limit provided in the COLR. ITS 3.2.3 Action F requires that the THERMAL POWER level of the unit be reduced to less than or equal to 20% RTP if QPT is greater than the maximum limit. Based on CTS 3.5.2.4.a and CTS 3.5.2.4.g, the QPT requirements are only applicable when operating at greater than 15% of rated power. Therefore, the CTS requirement to shut down the reactor (equivalent to ITS MODE 3) rather than to exit the Applicability (< 15% of rated power) presents requirements inconsistent with the requirements of CTS 3.0. ITS 3.2.3 is similarly applicable in MODE 1 with THERMAL POWER above 20% RTP (Refer to DOC L3 concerning difference between CTS 15% and ITS 20%). As such, the proposed change is considered administrative since it removes inconsistent statements within CTS 3.5.2.4 and clarifies that the required action when QPT is exceeded is to exit the applicability.

A11 CTS Specification 3.1.3.5 establishes the LCO requirements for regulating rod group positions in accordance with the position limits provided in the COLR. When position limits are exceeded, CTS Specification 3.5.2.5.c requires corrective measures be initiated immediately to achieve an acceptable control rod position and that the acceptable control rod position be achieved within 2 hours.

ITS 3.2.1, which provides comparable requirements, establishes a uniform approach to the Required Actions when regulating rod group position, sequence or required overlap is not within the limits. The Required Actions are based on: 1) regulating rod group inserted into an incorrect sequence or group overlap requirements not within the limits (ITS 3.2.1 Condition A); or 2) regulating rod group inserted into the restricted region or unacceptable region (ITS 3.2.1 Condition B). The ITS differentiates between the types of regulating rod group deviations, given above, that were not differentiated between in the CTS.

ITS ACTION A addresses those situations where the regulating rod group sequence or overlap requirements are not met. Required Action A.1 requires that the regulating rods be restored to within limits with a Completion Time of 2 hours. This is consistent with CTS 3.5.2.5.b. For the situation where the regulating rods are out of sequence, CTS 3.5.2.5.c implies (although does not explicitly state) that a completion time of 2 hours exists to restore compliance with the LCO. Therefore, the proposed change in format and presentation described above is considered administrative.

The ITS and CTS requirements are the same for situations in which the regulating rod groups are inserted into the restricted region or unacceptable region. The Completion Time for restoration of adherence to the limits is 2 hours from entry into ITS 3.2.1 Condition B. This maintains the present 2 hour restoration requirement established by CTS 3.5.2.5.c.

A12 CTS 3.1.11 requires that available shutdown margin (SDM) be > 1% $\Delta k/k$ during all system conditions except refueling. The CTS Bases for CTS 3.1.11 clarifies that during power operation and startup the SDM is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the position limits of the Core Operating Limits Report (COLR). In MODES 1 and 2, ITS ensures SDM is met by complying with ITS LCO 3.2.1, "Regulating Rod Position Limits," and LCO 3.1.5, "Safety Rod Position Limits," ITS LCO 3.1.1 requires that the SDM be within the limits specified in the COLR (> 1% $\Delta k/k$) during MODES 3, 4 and 5. Therefore, CTS 3.1.11 is marked as applying in MODES 1 and 2 as it relates to ITS LCO 3.2.1 (refer to markup for Section 3.2.1)and LCO 3.1.5 (refer to markup for 3.1.5) and in MODES 3, 4 and 5 as it relates to ITS 3.1.1 (refer to markup for 3.1.1). Since compliance with the SDM is ensured by complying with LCO 3.1.5 and 3.2.1, the actual SDM limit need not be specified "during all system conditions except refueling." As such, the proposed change is considered administrative and is consistent with the NUREG.

A13 CTS 3.5.2.5.c requires corrective measures be taken immediately to achieve an acceptable control rod position when the control rod position limits are exceed. An acceptable control rod position shall then be
2 attained within 2 hours. ITS 3.2.1, Required Actions B.2.1 and B.2.2 similarly require actions be completed in a specified time period but do not require they be initiated immediately. The proposed change is categorized as administrative due to the relatively short period of time to complete the required actions which would necessitate that the action begin without delay. The proposed change is consistent with the NUREG.

A14 CTS 3.5.2.4.b.1 provides an option of reducing the quadrant power tilt within 2 hours when the maximum positive quadrant power tilt exceeds the Steady State Limit but is less than or equal to the Transient Limit provided in the Core Operating Limits Report. ITS 3.2.3 does not provide this option since the option to restore to within limits always exist. CTS 3.5.2.1.b.1 is unnecessary and is deleted. Since the requirements are not changed, the proposed change is considered administrative and is consistent with the NUREG.

3.2 POWER DISTRIBUTION LIMITS (4)

3.2.1 Regulating Rod ~~Insertion~~ Position (4) Limits

CTS

LCO 3.2.1 Regulating rod ~~insertion~~ position (4), sequence, and overlap limits specified in the COLR.

3.1.3.5
3.5.2.5.b
3.5.2.5.c

APPLICABILITY: MODES 1 and 2.

3.1.3.5
3.5.2 Applic
3.1.11

Supp. 1

NOTE
This LCO ~~is not applicable while performing~~ SR 3.1.4.2.
Not required for any regulating rod ~~inserted~~ positioned to perform

TSTF 216

Supp 2

INSERT from next page (7)

ACTIONS

CONDITION	REQUIRED ACTION (7)	COMPLETION TIME
~~B.~~ A. (4) Regulating rod groups inserted in restricted ~~operational~~ region, or ~~sequence or overlap~~, or any combination, not met. (9)	~~A.1~~ B.2.1 Perform SR 3.2.5.1. or unacceptable AND A.2 Restore regulating rod groups to within ~~limits~~ acceptable region.	Once per 2 hours (7) 24 (4) hours from discovery of failure to meet the LCO
	OR	
B. Required Action and associated Completion Time of Condition A not met.	~~B.1~~ B.2.2 Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group ~~insertion~~ position limits. (4)	2 hours

3.5.2.5.c

3.5.2.5.c

(continued)

A. Regulating rod groups sequence or overlap requirements not met.	A.1 Restore regulating rod groups to within limits.	2 hours

3.5.2.5.b

(9)

CTS

Supp. 2

Move to previous page where indicated (7)

Position (4)

NOTE
Not applicable to regulating rod group positioned in the restricted region.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Regulating rod groups inserted in unacceptable operational region.	C.1 (B) Initiate boration to restore SDM to ≥ 1% Δk/k within the limits specified in the COLR. (D)	15 minutes
AND		
	C.2.1 Restore regulating rod groups to within restricted operating region.	2 hours
	OR	
	C.2.2 Reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group insertion limits.	2 hours
C ~~B~~. Required Action and associated Completion Time of Condition (A) not met.	C ~~D~~.1 Be in MODE 3.	12 ~~6~~ hours (13)

3.5.2.1
3.5.2.5.c
TSTF-009, Rev. 1

DOC A5
3.5.2.5.b

A or B (10)

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes made for clarity, preference or consistency with the Improved Technical Specifications (ITS) Writer's Guide.

3 The requirement/statement is deleted since it is not applicable to this facility. The following requirements are renumbered, where applicable, to reflect this deletion.

4 Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5 NUREG LCO 3.2.2 is not adopted in the ITS. ONS CTS do not have position limits for APSRs. APSR position is not assumed in UFSAR analyses. The following requirements are renumbered, where applicable, to reflect this deletion.

6 NUREG 3.2.4 Completion Times for Required Actions E.1 and F.1 are changed from 2 to 4 hours (ITS 3.2.3 Required Action E.1 and F.1). The change makes the Completion Time for Required Action F.1 (reduce power to $\leq$ 20% RTP) consistent with the completion time allowed by CTS 3.5.2.4.f for the condition where Quadrant Power Tilt exceeds the maximum limit. The 4 hour Completion Time provides a more reasonable time frame for performing the required power reduction to less than or equal to 20% RTP from full power conditions and is consistent with current technical specifications. The Completion Time for Required Action E.1 is changed to 4 hours for consistency with the Completion Time for Required Action F.1. This is considered appropriate since a power reduction from full power conditions could also be required if the Required Action of C.1 or D.1 is not met.

7 ONS currently does not have an NRC approved methodology that would allow the use of power peaking limits to verify that operation outside the LCO limits for Regulating Rod Position, AXIAL POWER IMBALANCE (API) or QUADRANT POWER TILT (QPT) do not violate the ECCS or DNB criteria. Therefore, the NUREG allowance that allows the use of power peaking verification when LCO limits are not met cannot be justified. As such, NUREG 3.2.1 Required Action A.1, NUREG 3.2.3 Required Action A.1, NUREG 3.2.4 Required Action A.1.1 are not included in the ONS ITS. The Completion Time for NUREG 3.2.1 Required Action A.1 (restoring the regulating rod groups to within limits) and NUREG 3.2.3 Required Action A.1 (restoring AXIAL POWER IMBALANCE to within limits) is changed to

2 hours consistent with CTS 3.5.2.5.c and 3.5.2.6, respectively. Due to the deletion of NUREG 3.2.1 Required Action A.1 and the modification of the Completion Time for NUREG 3.2.1 Required Action A.2 to eliminate "...from discovery of failure to meet the LCO" (to retain the CTS requirements for regulating rods not within position limits), NUREG 3.2.1 Condition A was modified to include the condition where regulating rods are inserted in the unacceptable region as well as those inserted in the restricted region. It was necessary to combine the two conditions (NUREG Conditions A and C) because the deletion of "...from discovery of failure to meet the LCO" would allow rotating between the
2 two conditions for an indefinite period of time. NUREG 3.2.1 Required
2 Actions C.2.1 and C.2.2 were deleted since ITS provides similar Required
2 Actions for the combined conditions. NUREG 3.2.1 Required Action C.1
2 was consolidated with NUREG 3.2.1 Required Actions A.2 and B.2 to become
2 ITS 3.2.1 Required Actions B.1, B.2.1, and B.2.2 respectively. ITS
2 3.2.1 Required Action B.1 is modified by a Note indicating that it is
2 not applicable to regulating rod groups positioned within the restricted
2 region.

Due to the deletion of Required Action A.1.1 of Specification 3.2.4, the second Completion Time of Required Action A.1.2.1 is no longer necessary and is deleted. NUREG Required Action A.1 for Specifications 3.2.1 and 3.2.3 allows additional time to restore the LCO limits provided power peaking factors are verified within limits (SR 3.2.5.1) once per 2 hours following entry into one of the LCO Conditions for these Specifications. NUREG Required Action A.1.1 for Specification 3.2.4 allows power peaking verification as an alternative to reducing THERMAL POWER and reducing the nuclear overpower trip setpoint. The NUREG provisions that allow use of the power peaking verification are considered appropriate since the LCO for Regulating Rod Insertion Limits, API operating limits, and QPT are used to ensure the core operates within the power peaking limits. However, ONS does not currently have an approved methodology for correlating the power peaking factors to these operating limits. The removal of this requirement maintains action requirements which are consistent with the CTS for similar situations. The Required Actions are renumbered appropriately.

Since the power peaking verification option is currently not a viable alternative at ONS, the adoption of NUREG 3.2.5 would be inappropriate and is not included in the ONS ITS.

8 Not used.

9 NUREG 3.2.1 Condition A is entered when the regulating rod groups are inserted into the restricted region, or sequence or overlap requirements are not met. ITS 3.2.1 provides separate ACTIONS for rods inserted into the restricted or unacceptable region and regulating rod groups not meeting sequence or overlap requirements. ITS 3.2.1 Condition B addresses regulating rod groups inserted in the restricted region or unacceptable region. ITS 3.2.1 Condition A addresses regulating rod

groups sequence and overlap requirements not met. This was necessary to eliminate any confusion regarding what restoration to within limits means when applied to regulating rods discovered in the restricted
2 region or the unacceptable region. The ITS Required Action B.2.1 provides a clear and concise requirement that the regulating rods must be restored to the acceptable region. ITS 3.2.1 ACTION A provides a specific Required Action to restore the regulating rod groups to within limits with a Completion Time of 2 hours. Two hours is based on CTS 3.5.2.5.b which allows two hours to initiate a reactor shutdown for regulating groups inserted out-of-sequence. This is considered a reasonable period of time for restoration based on operating experience.

10 The addition of a new Condition A for Specification 3.2.1 (described in JFD 7 and 9 above) and the reordering of NUREG 3.2.1 Condition A, require that NUREG Specification 3.2.1 Condition D be revised to read
2 "Required Actions and associated Completion Times of Conditions A or B not met" to ensure that appropriate actions are provided should
2 Conditions A or B not be satisfied. The appropriate action is to remove the unit from the LCO APPLICABILITY which is accomplished by having the unit proceed to MODE 3 with a Completion Time of 12 hours.

11 Not used.

2 12 Not used.

13 The current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 is retained in the ITS.

14 Not used.

15 Not used.

16 Not used.

17 NUREG LCO 3.2.4 Applicability for Quadrant Power Tilt Limits is MODE 1 with THERMAL POWER > 20% RTP. CTS 3.5.2.4.a Applicability for Quadrant Power Tilt Limits is during power operation above 15%. ITS LCO 3.2.3 for Quadrant Power Tilt Limits adopts the NUREG 3.2.4 Applicability which is slightly less restrictive than the CTS Applicability (Refer to the Discussion of Changes for this section for the justification for this less restrictive change).

BASES

ACTIONS

A.1 (continued)

that a rod insertion limit is ejected rod worth limited, then the ejected rod worth is no more limiting than the SDM based rod insertion limit in the core design (Ref. 8). Ejected rod worth limits are independently maintained by the Required Actions of Conditions A and C.

~~A.2~~ B.1, B.2.1, and B.2.2

Indefinite operation with the regulating rods inserted in the restricted region, or in violation of the group sequence or overlap limits, is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, reactivity limits may not be met and the abnormal regulating rod insertion or group configuration may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long term fuel depletion pattern. Therefore, power peaking monitoring is allowed for up to 24 hours after discovery of failure to meet the requirements of this LCO. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations, thereby limiting the potential for an adverse xenon redistribution.

~~B.1~~

~~If~~ the regulating rods cannot be restored within the acceptable operating limits shown on the figures in the COLR within the required Completion Time (i.e., Required Action A.2 not met), then the limits can be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion limits in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the plant systems. Operation for up to 2 hours more in the restricted region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions

(continued)

BASES

ACTIONS

B.1 (continued)

or configurations and limits the potential for an adverse xenon redistribution. (5)

Move to previous page where indicated (1)

Operation in the unacceptable region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit. Therefore, the RCS boron concentration must be increased to restore the regulating rod insertion to a value that preserves the SDM and ejected rod worth limits. The RCS boration must occur as described in Section B 3.1.1. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted operational region, which restores the minimum SDM capability and reduces the potential ejected rod worth to within its limit. (2)

Supp. 2

Required Action B.1 requires position

Required Action B.1 is modified by a Note indicating that it is not applicable to regulating rod groups positioned in the restricted region.

C.2.1

The required Completion Time of 2 hours from initial discovery of a regulating rod group in the unacceptable region until its restoration to within the restricted operating region shown on the figures in the COLR allows sufficient time for borated water to enter the RCS from the chemical addition and makeup systems, thereby allowing the regulating rods to be withdrawn to the restricted region. Operation in the restricted region for up to an additional 2 hours is reasonable, based on limiting the potential for an adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. (5)

(continued)

BASES

ACTIONS (continued)

C.2.2

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion limits in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the plant systems. Operation for up to 2 hours more in the restricted region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

(5)

Supp. 2

C.1

Required Action and associated Completion Time of Condition A or B are not met,

(5)

If the regulating rods cannot be restored to within the acceptable operating limits for the original THERMAL POWER or if the power reduction cannot be completed within the required Completion Time, then the reactor ~~is~~ must be placed in MODE 3, a MODE in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 12 ~~8~~ hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging plant systems.

(2) (2) (5)

Supp. 1

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

TSTF-110, R2

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours or 4 hours, depending on whether the CONTROL ROD drive sequence alarm is OPERABLE or not, is acceptable because little rod motion occurs in 4 hours due to fuel burnup, and the probability of a deviation occurring simultaneously with an inoperable sequence monitor in this relatively short time frame is low. Also, the Frequency

(12) (3)

(continued)

ITS Section 3.2

ID 79

Subject: Revised Spec 3.2.1, RA B.1 (was RA C.1 but relettered due to change described in ID 78) Completion Time to require boration within 15 minutes consistent with 3.1.1 RA A.1 CT

CONDITION	REQUIRED ACTION		COMPLETION TIME
2 B. Regulating rod groups positioned in restricted or unacceptable region.	B.1	--------NOTE-------- Not applicable to regulating rod groups positioned in the restricted region. ----------	
		Initiate boration to restore SDM to within the limits specified in the COLR.	15 minutes
	AND		
	B.2.1	Restore regulating rod groups to within acceptable region.	2 hours
	OR		
	B.2.2	Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group position limits.	2 hours
2 C. Required Action and associated Completion Time of Condition A 2 or B or not met.	C.1	Be in MODE 3.	12 hours

BASES

ACTIONS (continued)

A.1

Operation with the regulating rod groups out of sequence or with the group overlap limits exceeded may represent a condition beyond the assumptions used in the safety analyses. The design calculations assume no deviation in nominal overlap between regulating rod groups. However, small deviations in group overlap, as allowed by the COLR, may occur and would not cause significant differences in core reactivity, in power distribution, or rod worth, relative to the design calculations. Group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a predetermined order. The Completion Time of 2 hours is intended to restrict operation in this condition because of the potential severity associated with gross violations of group sequence or overlap requirements. The 2 hour Completion Time is based on operating experience which supports the restoration time without unnecessarily challenging unit operation and the low probability of an event occurring simultaneously with the limit out of specification.

2 B.1, B.2.1 and B.2.2

Operation in the unacceptable region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods
2 may be positioned too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit.
2 Therefore, Required Action B.1 requires the RCS boron concentration be increased to restore the regulating rod position to a value that preserves the SDM and ejected rod worth limits. The RCS boration must occur as described in
2 Section B 3.1.1. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted region, which restores the minimum SDM and reduces the potential ejected
2 rod worth to within its limit. Required Action B.1 is

(continued)

(A1) <except as marked>

3.2

~~3.5.2~~ ~~Control Rod Group and~~ Power Distribution Limits

3.2.1 App

Applicability

This specification applies to power distribution and operation of control rods during power operation. (A9) (M1)

MODES 1 & 2

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

Supp 2 ... RA B.1

~~3.5.2.1~~ Shutdown Margin

a. The available shutdown margin shall be greater than 1% Δk/k with the highest worth control rod fully withdrawn.

With Regulating Rods inserted in unacceptable region

b. If the shutdown margin is less than 1% Δk/k, then within ~~1 hour~~ initiate and continue boration until the required shutdown margin is restored. The requirements of Specification 3.5.2.5.c shall be met. (A4) 15 minutes (M7)

RA B.2.1

3.5.2.2 Movable Control Assemblies

a. All control (safety and regulating) rods shall be operable and positioned within nine (9) inches of their group average height.

b. A control rod shall be declared inoperable if any of the following conditions exist for that rod:

1. The control rod cannot be moved due to excessive friction or mechanical interference, or cannot perform its intended trip function.

2. The control rod cannot be located by either absolute or relative position indication or by in or out limit lights.

3. The control rod is misaligned with its group average by more than nine (9) inches.

4. The control rod does not meet the exercise requirements of Specification 4.1.

5. The control rod does not meet the rod trip insertion times of Specification 4.7.1.

6. The control rod does not meet the rod program verification of Specification 4.7.2.

(SEE 3.1)

A 66/66/63
12/15/78

not met entry into CTS 3.0 is required. Therefore the more limiting actions would be to initiate a shutdown per CTS 3.0.

CTS 3.5.2.1 modifies the allowable position limits of CTS 3.1.3.5 for the safety and regulating rod groups based on the ability to satisfy an imposed SHUTDOWN MARGIN requirement defined within CTS 3.5.2.1. With respect to the safety rod groups, this limitation requires that SDM be evaluated any time a safety rod is less than fully withdrawn during an approach to criticality, or for that matter, with the reactor critical (comparable to ITS MODES 1 and 2)(Ref. CTS 3.1.3.5). The requirement for the safety rod groups are retained in ITS 3.1.5 and are discussed in the Discussion of Changes (DOCs) for Section 3.1. With respect to the regulating rod groups, this SDM requirement is maintained in the ITS through: 1) the basis for derivation of the regulating rod group position limits given in the COLR, and 2) the Required Actions for
2 initiation of boration to restore SDM within 15 minutes (LCO 3.2.1
2 Required Action B.1) and the Required Actions to restore regulating rod
2 groups to within the acceptable region (LCO 3.2.1 Required Action B.2.1) or reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group's position limits (LCO 3.2.1
2 Required Action B.2.2). This change represents a change in format only. Requirements consistent with the referenced portion of the CTS are maintained by the adoption of this ITS requirement. No additional requirements are added and no existing requirements are removed by this change. The proposed change is consistent with the NUREG with the exception to the modifications described in JFDs for the NUREG Specification 3.2.1.

A5 The CTS currently does not have a shutdown requirement specific to CTS 3.5.2.5.c when the requirements of 3.5.2.5.c (i.e., attaining an acceptable control rod position) are not met. Therefore, CTS 3.0
2 requirements apply. ITS ACTION C is added to provide an explicit requirement to be in MODE 3 within 12 hours of failure to restore sequence or position limits. This is equivalent to the shutdown time in CTS 3.0. Therefore, the proposed change is considered administrative and is consistent with the NUREG.

A6 CTS 3.5.2.5.c states that the regulating control rod position limits shall be maintained within acceptable operating limits for regulating control rod position provided in the COLR for the particular number of operating reactor coolant pumps (4,3). These limits are already in the COLR based on the number of operating reactor coolant pumps (RCP). Therefore, specific reference to the number of operating RCPs provides no requirement and has been deleted. Therefore, the proposed change is considered administrative and is consistent with the NUREG.

A7 CTS 3.5.2.4.b.2 for Quadrant Power Tilt (QPT) differentiates action to be taken when the maximum QPT exceeds the steady state limit but is less than or equal to the transient limit provided in the COLR based on reducing from 100% power or reducing from the allowable thermal power for the reactor coolant pump combination. Comparable ITS LCO 3.2.3

M4 CTS does not specify a requirement to periodically verify rod sequence and overlap limits and rod position limits are met. ITS Surveillance Requirements for rod sequence and overlap limits and rod position limits are added (ITS SRs 3.2.1.1 and 3.2.1.2, respectively). SR 3.2.1.1 ensures that the regulating rod groups sequence and overlap limits are not violated. SR 3.2.1.2, which verifies regulating rod groups position limits, is intended to detect regulating rod banks that may be approaching their position limits. These requirements are consistent with existing limiting conditions for operation in CTS 3.5.2, however the CTS does not specify a requirement to periodically verify these parameters are within limits. Therefore, this change represents an appropriate yet more restrictive requirement on plant operation. This change is consistent with the NUREG.

M5 CTS 3.5.2.4.f requires the reactor be shutdown if the maximum positive quadrant power tilt exceeds the maximum limit in the COLR. CTS permits subsequent reactor operation for the purpose of measurement, testing, and corrective action provided the thermal power and nuclear overpower trip setpoints are restricted by a reduction of 2% of thermal power for each 1% of tilt for the maximum tilt observed prior to shutdown. For the same condition, ITS 3.2.3 Required Action F.1 requires THERMAL POWER to be reduced to $\leq$ 20% RTP. This is more restrictive since CTS 3.5.2.4.f allows the unit to operate at thermal power levels up to 60% RTP for the purposes previously described. ITS 3.2.3 Required Action F.1 is considered appropriate since QPT in excess of the maximum limit can be an indication of a severe power distribution anomaly, and a power reduction to at most 20% RTP ensures local linear heat rates do not exceed allowable limits while the cause is being determined and corrected. This change is consistent with the NUREG.

M6 CTS 3.1.11 requires required shutdown margin (SDM) to be met during all system conditions. CTS 3.5.2.1 requires the SDM to be met during power operation. However, the CTS does not specifically require SDM to be verified. ITS SR 3.2.1.3 is added to require SDM be verified within 4 hours prior to achieving criticality. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shutdown or trip the reactor after criticality. This requirement is consistent with existing limiting conditions for operation in CTS 3.5.2 and CTS 3.1.11, however the CTS does not specify a requirement to verify SDM. Therefore, this change represents an appropriate yet more restrictive requirement on plant operation. This change is consistent with the NUREG.

2 M7 CTS 3.5.2.1.b requires boration be initiated within one hour and
2 continued until required shutdown margin is restored when not met. ITS
2 3.2.1 Required Action B.1 requires boration be initiated within 15
2 minutes. The change represents a reasonable yet more restrictive
2 requirement on plant operation. This change is consistent with the
2 NUREG.

CTS

Supp. 2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Regulating rod groups inserted in unacceptable operational region.	C.1 Initiate boration to restore SDM to ≥ 1% Δk/k within the limits specified in the COLR. AND	15 minutes
	C.2.1 Restore regulating rod groups to within restricted operating region. OR	2 hours
	C.2.2 Reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group insertion limits.	2 hours
C. ~~B~~. Required Action and associated Completion Time of Condition C not met.	C. ~~B~~.1 Be in MODE 3.	12 ~~6~~ hours

Move to previous page where indicating (7)

NOTE — Not applicable to regulating rod group positioned in the restricted region.

(4) Position ~~Insertion~~

(1)

(7)

TSTF-009, Rev. 1

3.5.2.1
3.5.2.5.c

(13)

DOC A5
3.5.2.5.b

(11) A or B (10)

groups sequence and overlap requirements not met. This was necessary to eliminate any confusion regarding what restoration to within limits means when applied to regulating rods discovered in the restricted
2 region or the unacceptable region. The ITS Required Action B.2.1 provides a clear and concise requirement that the regulating rods must be restored to the acceptable region. ITS 3.2.1 ACTION A provides a specific Required Action to restore the regulating rod groups to within limits with a Completion Time of 2 hours. Two hours is based on CTS 3.5.2.5.b which allows two hours to initiate a reactor shutdown for regulating groups inserted out-of-sequence. This is considered a reasonable period of time for restoration based on operating experience.

10 The addition of a new Condition A for Specification 3.2.1 (described in JFD 7 and 9 above) and the reordering of NUREG 3.2.1 Condition A, require that NUREG Specification 3.2.1 Condition D be revised to read
2 "Required Actions and associated Completion Times of Conditions A or B not met" to ensure that appropriate actions are provided should
2 Conditions A or B not be satisfied. The appropriate action is to remove the unit from the LCO APPLICABILITY which is accomplished by having the unit proceed to MODE 3 with a Completion Time of 12 hours.

11 Not used.

2 12 Not used.

13 The current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 is retained in the ITS.

14 Not used.

15 Not used.

16 Not used.

17 NUREG LCO 3.2.4 Applicability for Quadrant Power Tilt Limits is MODE 1 with THERMAL POWER > 20% RTP. CTS 3.5.2.4.a Applicability for Quadrant Power Tilt Limits is during power operation above 15%. ITS LCO 3.2.3 for Quadrant Power Tilt Limits adopts the NUREG 3.2.4 Applicability which is slightly less restrictive than the CTS Applicability (Refer to the Discussion of Changes for this section for the justification for this less restrictive change).

BASES

ACTIONS

B.1 (continued)

or configurations and limits the potential for an adverse xenon redistribution. ⑤

Move to Previous page where indicated

B.1

Operation in the unacceptable region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit. Therefore, the RCS boron concentration must be increased to restore the regulating rod insertion to a value that preserves the SDM and ejected rod worth limits. The RCS boration must occur as described in Section B 3.1.1. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted operational region, which restores the minimum SDM capability and reduces the potential ejected rod worth to within its limit. ②

Supp. 2

Required Action B.1 requires

position

Required Action B.1 is modified by a Note indicating that it is not applicable to regulating rod groups positioned in the restricted region.

C.2.1

The required Completion Time of 2 hours from initial discovery of a regulating rod group in the unacceptable region until its restoration to within the restricted operating region shown on the figures in the COLR allows sufficient time for borated water to enter the RCS from the chemical addition and makeup systems, thereby allowing the regulating rods to be withdrawn to the restricted region. Operation in the restricted region for up to an additional 2 hours is reasonable, based on limiting the potential for an adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. ⑤

(continued)

ITS Section 3.2

ID 81

Subject: Revised 3.2.2 to change "reduce" to "restore" AXIAL POWER IMBALANCE to within limits.

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.2 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
2 A. AXIAL POWER IMBALANCE not within limits.	A.1 Restore AXIAL POWER IMBALANCE to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 40% RTP.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	12 hours

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2 (3.2.2) AXIAL POWER IMBALANCE Operating Limits

LCO 3.2 (3.2.2) AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR. (5) 3.5.2.6

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP. 3.5.2.6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. AXIAL POWER IMBALANCE not within limits.	A.1 Perform SR 3.2.5.1.	Once per 2 hours (7)	
	AND		
	A.2 Reduce AXIAL POWER IMBALANCE within limits. (10) Restore	24 hours (2)	3.5.2.6
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 40% RTP.	2 hours	3.5.2.6

Supp 2

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

A1 <except as marked>

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

~~Channel Description~~ Function	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	<SEE 3.3.3 + 4>
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
1.a~~3~~. Power Range Amplifier SR 3.3.1.2	~~ES~~(1) 24 hrs (L9)	NA	(1)	(1) Heat balance check each shift (24 hrs) (L9) Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
Supp 1 1.a~~4~~. Power Range Nuclear Overpower 1.b 8 Nuclear Overpower Flux/Flow Imbalance (A5)	~~ES~~ 12 hrs SR 3.3.1.1 (A9)	45 Days STB SR 3.3.1.4 (A9)	~~MO~~(1)(2) 31 days SR 3.3.1.3 (For Funct. 8 only) (A9) For Funct. 1.a + 8 only	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week. (L8) SR 3.3.1.3
5. Wide Range	ES(1)	PS	NA	(1) When in service. <SEE 3.3.8, 9 + 10>
6. Source Range	ES(1)	PS	NA	(1) When in service. <SEE 3.3.8, 9, 10, 3.9.2>
Page 8 of 10 2 ~~7~~. Reactor Coolant Temperature Supp 11	SR 3.3.1.1 ~~ES~~	SR 3.3.1.4 45 Days STB (A9)	SR 3.3.1.5 18 months	
3 11 ~~8~~. High Reactor Coolant Pressure Supp 1	~~ES~~ 12 hrs	45 Days STB	18 months	
4 ~~9~~. Low Reactor Coolant Pressure Supp 2	~~ES~~	45 Days STB	18 months	

Add SR 3.3.1.2 Note (L14) !
Add SR 3.3.1.3 Note (L15) !
Add SR 3.3.1.5 Note for Functions 2, 3, 4, 5, 6, 7, 8, 9, 10 + 11 (L20) !
Add SR 3.3.1.5 for Functions 1.a + 1.b (M35)

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Table 4.1-1 (CONTINUED)

SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5

	Channel Description	Check	Test	Calibrate	Remarks
8 ~~10.~~	Flux-Reactor Coolant Flow Comparator	~~ES~~ 12 hrs	45 Days STB	18 months	
5 ~~11.~~	Reactor Coolant Pressure Temperature Comparator	~~ES~~ 12 hrs	45 Days STB	18 months	
7 ~~12.~~	Pump-Flux Comparator	~~ES~~ 12 hrs	45 Days STB	18 months	
6 ~~13.~~	High Reactor Building Pressure	~~DA~~ 12 hrs	45 Days STB	18 months	
14.	High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
15.	High Pressure Injection Analog Channels:				
a.	Reactor Coolant Pressure	ES	MO	18 months	
b.	Reactor Building Pressure (4 psig)	ES	MO	18 months	
16.	Low Pressure Injection Logic	NA	MO	NA	

A9

M17

<SEE 3.3.5+7>

Supp 2

Supp 2

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Specification 3.3.1

Amendment No.228(Unit 1)
Amendment No.229(Unit 2)
Amendment No.225(Unit 3)

~~Table 4.1-1 (CONTINUED)~~ (A1 <except as marked>)

Channel Description		Check	Test	Calibrate	Remarks
44.	Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	18 months	NA	
45.	Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	18 months(1)	NA	Includes Reactor Building isolation of essential systems only. (1) A one-time extension of the test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.
46.	Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	18 months(1)	NA	Includes Reactor Building isolation of essential systems only. (1) A one-time extension of the test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.
47.	Engineered Safeguards Channel 7 Spray Manual Trip	NA	18 months	NA	<SEE 3.3.6>
48.	Engineered Safeguards Channel 8 Spray Manual Trip	NA	18 months	NA	
49.	Emergency Feedwater Flow Indicators	MO	NA	18 months	<SEE 3.3.8>
50.	PORV and Safety Valve Position Indicators	MO	NA	18 months	R1
9 ~~51.~~	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	~~NA~~	SR 3.3.1.4 45 Days STB A91	SR 3.3.1.5 18 months	

Supp 2

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Specification 3.3.1

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

3.3.1-1 (A1) <except as marked>

Table ~~4.1-1~~ (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
10 ~~52.~~ RPS Anticipatory Reactor Trip System Loss of Main Feedwater		SR 3.3.1.4	SR 3.3.1.5	
Supp 2 ~~a)~~ Control Oil Pressure Switches	NA	45 Days STB A9	18 months	
53. Emergency Feedwater Initiation Circuits				<SEE 3.3.14>
a) Control Oil Pressure Switches	NA	MO	18 months	
54. Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	18 months	TMI Item II.F.1.3 <SEE 3.3.8>
55. Containment Pressure Monitor (PT-230, 231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	18 months	TMI Item II.F.1.5
57. Containment Hydrogen Monitor (MT-80,-81)	NA	MO	AN	TMI Item II.F.1.6
58. Wide Range Hot Leg Level	NA	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
59. Reactor Vessel Head Level	NA	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.

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Specification 3.3.1

Amendment No. 228(Unit 1)
Amendment No. 229(Unit 2)
Amendment No. 225(Unit 3)

~~Table 4.1-1 (CONTINUED)~~ (A1) <except as marked>

~~Channel Description~~		~~Check~~	~~Test~~	Calibrate	~~Remarks~~	
33.	Containment Temperature	NA	NA	18 months		<SEE 3.3.8>
34.	Incore Neutron Detectors	MO(1)	NA	NA	(1)	Check functioning; including functioning of computer readout or recorder readout. <SEE 3.2>
35.	Emergency Plant Radiation Instruments	MO(1)	NA	18 months	(1)	Battery check.
36.	Environmental Monitors	MO(1)	NA	18 months	(1)	Check functioning.
37.	Reactor Manual Trip	~~NA~~	PS	~~NA~~ SR 3.3.2.1		
38.	Reactor Building Emergency Sump Level	NA	NA	18 months		(A9) SR 3.3.2.1 Freq
39.	Steam Generator Water Level	WE	NA	18 months		
40.	Turbine Overspeed Trip	NA	NA	18 months		
41.	Engineered Safeguards Channel 1 HP Injection & Reactor Building Isolation Manual Trip	NA	18 months	NA		Includes Reactor Building isolation of non-essential systems only
42.	Engineered Safeguards Channel 2 HP Injection & Reactor Building Isolation Manual Trip	NA	18 months	NA		Includes Reactor Building isolation of non-essential systems only
43.	Engineered Safeguards Channel 3 LP Injection Manual Trip	NA	18 months	NA		<SEE 3.3.6>

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Specification 3.3.2

Amendment No. 228(Unit 1)
Amendment No. 229(Unit 2)
Amendment No. 225(Unit 3)

Table 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate		Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	~~MO~~ 31 days (SR 3.3.3.1) (A9)	NA		
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1)	This test shall independently confirm the operability of the shunt trip device and the undervoltage device. (SEE 3.3.4)
3. Power Range Amplifier	ES(1)	NA	(1)	(1)	Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) (2)	Using incore instrumentation. Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide Range	ES(1)	PS	NA	(1)	When in service. (SEE 3.3.8, 9 + 10, 3.9.2)
6. Source Range	ES(1)	PS	NA	(1)	When in service.
7. Reactor Coolant Temperature	ES	45 Days STB	18 months		1
8. High Reactor Coolant Pressure	ES	45 Days STB	18 months		1
9. Low Reactor Coolant Pressure	ES	45 Days STB	18 months		1 (SEE 3.3.1)

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Specification 3.3.3

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Table 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	<SEE 3.3.3>
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	~~MO~~ SR 3.3.4.1 31 days	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide Range	ES(1)	PS	NA	(1) When in service.
6. Source Range	ES(1)	PS	NA	(1) When in service.
7. Reactor Coolant Temperature	ES	45 Days STB	18 months	
8. High Reactor Coolant Pressure	ES	45 Days STB	18 months	
9. Low Reactor Coolant Pressure	ES	45 Days STB	18 months	

<SEE 3.3.8, 9, 10 & 3.9.2>

<SEE 3.3.1>

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Specification 3.3.4

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Table 4.1-1 (CONTINUED)

All (except as marked)

Channel Description	Check	Test	Calibrate	Remarks
10. Flux-Reactor Coolant Flow Comparator	ES	45 Days STB	18 months	
11. Reactor Coolant Pressure Temperature Comparator	ES	45 Days STB	18 months	(SEE 3.3.1)
12. Pump-Flux Comparator	ES	45 Days STB	18 months	
13. High Reactor Building Pressure	DA	45 Days STB	18 months	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems (SEE 3.3.7)
15. High Pressure Injection Analog Channels:				
Parameter	SR 3.3.5.1	SR 3.3.5.2	SR 3.3.5.3	Add SR 3.3.5.2 Note (L6)
1 ~~a.~~ Reactor Coolant Pressure	~~ES~~ 12 hrs ~~ES~~	~~MO~~ 31 days ~~MO~~	18 months	
3 ~~b.~~ Reactor Building Pressure (4 psig)			18 months	
16. Low Pressure Injection Logic	NA	MO	NA	(SEE 3.3.7)

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Specification 3.3.5

Amendment No.228 (Unit 1)
Amendment No.229 (Unit 2)
Amendment No.225 (Unit 3)

TABLE 4.1-1 (CONTINUED)

A1 <except as marked>

Channel Description	Check	Test	Calibrate	Remarks
17. Low Pressure Injection Analog Channels:		SR 3.3.5.1	SR 3.3.5.2	SR 3.3.5.3
Supp 2 | 2 ~~a.~~ Reactor Coolant Pressure		~~ES~~ 12 hrs	~~MO~~ 31 days	18 months
3 ~~b.~~ Reactor Building Pressure (4 psig)		~~ES~~ A9	~~MO~~ A9	18 months
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems <SEE 3.3.7>
Supp 2 | 3 ~~19.~~ Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)		~~ES~~ 12 hrs A9	~~MO~~ 31 days A9	18 months
20. Reactor Building Spray System Logic	NA	MO	NA	<SEE 3.3.7>
Supp 2 | 21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure		~~NA~~ Add SR 3.3.5.1 M21	~~MO~~ 31 days A9	18 months
Pen & Ink 22. Pressurizer Temperature	ES	NA	18 months	R1
23. Control Rod Absolute Position	ES(1)	NA	18 months (2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel. <SEE 3.1>

Specification 3.3.5

Amendment No. 228(Unit 1)
Amendment No. 229(Unit 2)
Amendment No. 225(Unit 3)

Table 4.1-1 (CONTINUED) (A1 except as marked)

Channel Description	Check	Test	Calibrate	Remarks
33. Containment Temperature	NA	NA	18 months	(SEE 3.3.8)
34. Incore Neutron Detectors	MO(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout. (SEE 3.2)
35. Emergency Plant Radiation Instruments	MO(1)	NA	18 months	(1) Battery check.
36. Environmental Monitors	MO(1)	NA	18 months	(1) Check functioning.
37. Reactor Manual Trip	NA	PS	NA	(SEE 3.3.2)
38. Reactor Building Emergency Sump Level	NA	NA	18 months	(SEE 3.3.8)
39. Steam Generator Water Level	WE	NA	18 months	
40. Turbine Overspeed Trip	NA	NA	18 months	
41. Engineered Safeguards Channel 1 HP Injection & Reactor Building Isolation Manual Trip	~~NA~~	SR 3.3.6.1 18 months	~~NA~~	Includes Reactor Building isolation of non-essential systems only (LA6)
42. Engineered Safeguards Channel 2 HP Injection & Reactor Building Isolation Manual Trip	~~NA~~	18 months	~~NA~~	Includes Reactor Building isolation of non-essential systems only
43. Engineered Safeguards Channel 3 LP Injection Manual Trip	~~NA~~	18 months	~~NA~~	

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Specification 3.3.6

Amendment No. 228(Unit 1)
Amendment No. 229(Unit 2)
Amendment No. 225(Unit 3)

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
44. Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	18 months	NA	
45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	18 months (1)	NA	Includes Reactor Building isolation of essential systems only. (1) A one-time extension of the test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.
46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	18 months (1)	NA	Includes Reactor Building isolation of essential systems only. (1) A one-time extension of the test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.
47. Engineered Safeguards Channel 7 Spray Manual Trip	NA	18 months	NA	
48. Engineered Safeguards Channel 8 Spray Manual Trip	NA	18 months	NA	
49. Emergency Feedwater Flow Indicators	MO	NA	18 months	SEE 3.3.8
50. PORV and Safety Valve Position Indicators	MO	NA	18 months	SEE 3.3.1
51. RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	18 months	SEE 3.3.1

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
10. Flux-Reactor Coolant Flow Comparator	ES	45 Days STB	18 months	
11. Reactor Coolant Pressure Temperature Comparator	ES	45 Days STB	18 months	
12. Pump-Flux Comparator	ES	45 Days STB	18 months	
13. High Reactor Building Pressure	DA	45 Days STB	18 months	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	~~NA~~	~~MO~~	NA	Includes Reactor Building Isolation of non-essential systems
15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	18 months	
b. Reactor Building Pressure (4 psig)	ES	MO	18 months	
16. Low Pressure Injection Logic	~~NA~~	~~MO~~	NA	

A1

SEE 3.3.1

SR 3.3.7.1

31 days

A9

LA6

SEE 3.3.5

SR 3.3.7.1

31 days

A9

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Specification 3.3.7

Amendment No.228(Unit 1)
Amendment No.229(Unit 2)
Amendment No.225(Unit 3)

TABLE 4.1-1 (CONTINUED)

Channel Description		Check	Test	Calibrate	Remarks
17.	Low Pressure Injection Analog Channels:				
	a. Reactor Coolant Pressure	ES	MO	18 months	SEE 3.3.5
	b. Reactor Building Pressure (4 psig)	ES	MO	18 months	
18.	Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	~~NA~~ SR 3.3.7.1	~~MO~~ 31 days	NA	Reactor Building isolation includes essential systems
19.	Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	18 months	SEE 3.3.5
20.	Reactor Building Spray System Logic	~~NA~~	~~MO~~ 31 days	NA	
21.	Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	18 months	SEE 3.3.5
22.	Pressurizer Temperature	ES	NA	18 months	
23.	Control Rod Absolute Position	ES(1)	NA	18 months (2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel. SEE 3.1

except as marked

Supp 2

Supp 2

Supp 2

Pgs 8 & 9

Specification 3.3.7

Add SR Note (A19)

(except as marked)

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

(SEE 3.3.3 + 4)

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
~~5.~~ Wide Range	31 days ~~ES~~(1)	PS	NA	(1) When in service.
6. Source Range	ES(1)	PS	NA	(1) When in service.
7. Reactor Coolant Temperature	ES	45 Days STB	18 months	
8. High Reactor Coolant Pressure	ES	45 Days STB	18 months	
9. Low Reactor Coolant Pressure	ES	45 Days STB	18 months	

(SEE 3.3.1)

(SEE 3.3.1)

L26 → SR 3.3.8.1

(SEE 3.3.9, 10 + 3.9.2)

Add SR 3.3.8.3 for Function 1 (M29)

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Specification 3.3.8

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

(A1 except as marked)

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
24. Control Rod Relative Position	ES(1)	NA	18 months (2)	(SEE 3.1) (1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel.
25. Core Flood Tanks				
a. Pressure	ES	NA	18 months	R1
b. Level	ES	NA	18 months	R1
11 ~~26.~~ Pressurizer Level	SR 3.3.8.1 ~~ES~~ 31 days	~~NA~~ L26	18 months - SR 3.3.8.3	
27. Letdown Storage Tank Level	DA	NA	18 months	R1
28. Delete				
~~29.~~ 18 + 19 High and Low Pressure Injection Systems Flow Channels	~~NA~~ Add SR 3.3.8.1 M26	~~NA~~ L26	18 months - SR 3.3.8.3	
14 ~~30.~~ Borated Water Storage Tank Level Indicator	~~WE~~ 31 days	~~NA~~	18 months - SR 3.3.8.3	
31. Boric Acid Mix Tank:				
a. Level	NA	NA	AN	R1
b. Temperature	MO	NA	AN	R1
32. Concentrated Boric Acid Storage Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	

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Specification 3.3.8

~~Table 4.1-1 (CONTINUED)~~ (A1) <except as marked>

Channel Description		Check	Test	Calibrate	Remarks
44.	Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	18 months	NA	
45.	Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	18 months(1)	NA	Includes Reactor Building isolation of essential systems only. (1) A one-time extension of the test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.
46.	Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	18 months(1)	NA	Includes Reactor Building isolation of essential systems only. (1) A one-time extension of the test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16.
47.	Engineered Safeguards Channel 7 Spray Manual Trip	NA	18 months	NA	<SEE 3.3.6>
48.	Engineered Safeguards Channel 8 Spray Manual Trip	NA	18 months	NA	
2| ~~49.~~	Emergency Feedwater Flow Indicators	~~MO~~ 31 days (A9) SR 3.3.8.1	NA	18 months SR 3.3.8.3	
50.	PORV and Safety Valve Position Indicators	MO	NA	18 months	<SEE 3.3.1>
51.	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	18 months	

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1 & 2 of 3 Amd

Specification 3.3.8

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

A1 except as marked

~~Table 4.1-1 (CONTINUED)~~

~~Channel Description~~	~~Check~~	~~Test~~	~~Calibrate~~	~~Remarks~~
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater				SEE 3.3.1
Supp 2 a) Control Oil Pressure Switches	NA	45 Days STB	18 months	
53. Emergency Feedwater Initiation Circuits				SEE 3.3.14
Supp 2 a) Control Oil Pressure Switches	NA	MO	18 months	
Supp 2 9 ~~54.~~ Containment High Range Radiation Monitor (RIA-57, 58)	SR 3.3.8.1 ~~NA~~ 31 days	~~MO~~ M33	SR 3.3.8.2/3 18 months	TMI Item II.F.1.3 A24
7 ~~55.~~ Containment Pressure Monitor (PT-230, 231)	~~MO~~ 31 days A9	~~NA~~	AN 12 months A9	TMI Item II.F.1.4
Supp 2 6 ~~56.~~ Containment Water Level Monitor-Wide Range (LT-90, -91)	~~MO~~	~~NA~~	18 months	TMI Item II.F.1.5
10 ~~57.~~ Containment Hydrogen Monitor (MT-80, 81)	~~NA~~ 31 days L27	MO	AN 12 months A9	TMI Item II.F.1.6
Supp 2 3 ~~58.~~ Wide Range Hot Leg Level	~~NA~~ 31 days	18 months (1)	18 months (1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
Supp 2 5 ~~59.~~ Reactor Vessel Head Level	~~NA~~ 31 days M33	18 months (1) A22	18 months (1) A37	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.

LAS

Amendment No. 228(Unit 1)
Amendment No. 229(Unit 2)
Amendment No. 225(Unit 3)

Specification 3.3.8

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(A1 except as marked)

~~Table 4.1-1 (CONTINUED)~~

	~~Channel Description~~	SR 3.3.8.1 ~~Check~~	~~Test~~	SR 3.3.8.3 ~~Calibrate~~	~~Remarks~~
Supp 2	16 ~~60.~~ Core Exit Thermocouples	~~MO~~ 31 days (A9)	NA	18 months (1)	(1) A one-time extension of the calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16. (A37)
	17 ~~61.~~ Subcooling Monitors	~~MO~~ 31 days (A9)	18 months (1) (A37)	18 months (1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
	62. Main Steam Header Pressure and MSLB detection (analog) channels	ES	18 months	18 months	(SEE 3.3.11)
	63. Feedwater isolation circuitry (digital) channels and manual pushbutton	NA	18 months	NA	(SEE 3.3.12 + 13)

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
STB - STAGGERED TEST BASIS
(A9)

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Specification 3.3.8

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

Table 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide Range	ES(1)	PS	NA	(1) When in service.
6. Source Range	~~ES(1)~~ 12 hrs	~~PS~~	~~NA~~	~~(1) When in service.~~
7. Reactor Coolant Temperature	ES	45 Days STB	18 months	
8. High Reactor Coolant Pressure	ES	45 Days STB	18 months	
9. Low Reactor Coolant Pressure	ES	45 Days STB	18 months	

SEE 3.3.3 + 3.3.4

SEE 3.3.1

SEE 3.3.8 + 10

SR 3.3.9.1

A9

A1

L32

Add SR 3.3.9.2

M14

SEE 3.3.1

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Specification 3.3.9

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

(A1) except as marked)

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

(SEE 3.3.3 + 3.3.4)

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(SEE 3.3.1) (1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide Range	(A4) ~~ES(1)~~ 12 hrs	(L32) PS	(Add SR 3.3.10.2) (M14) NA	(1) ~~When in service.~~ (A1)
6. Source Range	ES(1)	PS	NA	(1) When in service. (SEE 3.3.9)
7. Reactor Coolant Temperature	ES	45 Days STB	18 months	(SEE 3.3.1)
8. High Reactor Coolant Pressure	ES	45 Days STB	18 months	
9. Low Reactor Coolant Pressure	ES	45 Days STB	18 months	

SR 3.3.10.1

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Specification 3.3.10

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Table 4.1-1 (CONTINUED)

SEE 3.3.8

	Channel Description	Check	Test	Calibrate	Remarks
Supp 2	60. Core Exit Thermocouples	MO	NA	18 months(1)	(1) A one-time extension of the calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
	61. Subcooling Monitors	MO	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
Supp 2	~~62.~~ Main Steam Header Pressure and MSLB detection (analog) channels	~~ES~~ 12 hrs (A9)	SR 3.3.11.1	~~18 months~~ (A15)	18 months
	63. Feedwater isolation circuitry (digital) channels and manual pushbutton	NA	18 months	NA	<SEE 3.3.12 + 13>

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
STB - STAGGERED TEST BASIS

Supp 2 — (A9)

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Specification 3.3.11

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

(A1) (except as marked)

(SEE 3.3.8)

Table 4.1-1 (CONTINUED)

	Channel Description	Check	Test	Calibrate	Remarks
Supp 2	60. Core Exit Thermocouples	MO	NA	18 months(1)	(1) A one-time extension of the calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
	61. Subcooling Monitors	MO	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
Supp 2	62. Main Steam Header Pressure and MSLB detection (analog) channels	ES	18 months	18 months	(SEE 3.3.11)
			SR 3.3.12.1		
Supp 2	63. Feedwater isolation circuitry (digital) channels and manual pushbutton (SEE 3.3.13)	~~NA~~	18 months	~~NA~~	

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
STB - STAGGERED TEST BASIS

(A9)

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Specification 3.3.12

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

Table 4.1-1 (CONTINUED)

(SEE 3.3.8)

Channel Description		Check	Test	Calibrate	Remarks
Supp 2 60.	Core Exit Thermocouples	MO	NA	18 months(1)	(1) A one-time extension of the calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
61.	Subcooling Monitors	MO	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
Supp 2 62.	Main Steam Header Pressure and MSLB detection (analog) channels	ES	18 months	18 months	(SEE 3.3.11)
Supp 2 63.	Feedwater isolation circuitry (digital) channels and manual pushbutton	NA	18 months	NA	(SEE 3.3.12)

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ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
STB - STAGGERED TEST BASIS

Supp 2 (A9)

Specification 3.3.13

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

~~Table 4.1-1 (CONTINUED)~~ A1 <except as marked>

Channel Description	Check	Test	Calibrate	Remarks
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater				SEE 3.3.1
a) Control Oil Pressure Switches	NA	45 Days STB	18 months	
53. Emergency Feedwater Initiation Circuits		SR 3.3.14.1	SR 3.3.14.2	M20 Add 3.3.14.1 for manual initiation circuit
a) Control Oil Pressure Switches	NA	~~MO~~ 31 days A9	18 months	
54. Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	18 months	TMI Item II.F.1.3 <SEE 3.3.8>
55. Containment Pressure Monitor (PT-230, 231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	18 months	TMI Item II.F.1.5
57. Containment Hydrogen Monitor (MT-80,-81)	NA	MO	AN	TMI Item II.F.1.6
58. Wide Range Hot Leg Level	NA	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
59. Reactor Vessel Head Level	NA	18 months(1)	18 months(1)	(1) A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.

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Specification 3.3.14

Amendment No. 228(Unit 1)
Amendment No. 229(Unit 2)
Amendment No. 225(Unit 3)

Specification 3.3.16

(A1 <except as marked>)

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency	
1.	Control Rod Movement (1)	Movement of Each Rod	Monthly	<SEE 3.1>
2.	Pressurizer Safety Valves	Setpoint	18 months(4)	<SEE 3.4>
3.	Main Steam Safety Valves	Setpoint	18 months(4)	<SEE 3.7>
4.	Refueling System Interlocks(5)	Functional	Prior to Refueling	L35
5.	Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly	<SEE 3.7>
6.	Reactor Coolant System Leakage (2)	Evaluate	Daily	
7.	Emergency Condenser Circulating Water System Test (6)	Functional	18 months	
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	<SEE 3.7>
9.	Spent Fuel Cooling System	Functional	Prior to Refueling	
10.	High Pressure and Low Pressure Injection System (3)	Vent Pump Casings	Monthly and Prior to Testing	<SEE 3.5>
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	<SEE 3.7>
12.	MSLB Feedwater Isolation Feature (7)	Functional	18 months	
13.	Essential Siphon Vacuum System Test (8)	Functional	Quarterly	

Supp 2

SR 3.3.16.2

Supp 2

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Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.3.16

(1) Applicable only when the reactor is critical. <SEE 3.1 + 3.7>

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded. — <SEE 3.5>

<SEE 3.4 + 3.7>

Supp 2

(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

SR 3.3.16.2

(5) Applicable only to the interlocks associated with the Reactor Building Purge System.

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

<SEE 3.7>

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature.

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit.

including Table 3.3.1-1, deals individually with each RPS trip function which receives input from the power range instrument channels, including the nuclear overpower trip function. The power range nuclear instrument channels are not addressed separately by ITS. This is a change in the presentation of these requirements, with no actual change in requirements and is, therefore, administrative. The change is consistent with the NUREG.

A6 CTS 3.5.1.1 specifies that Table 3.5.1-1 Column C requirements be met for each functional unit. The required action when the minimum operable channel requirements of Column C are not met are prescribed separately for each functional unit. The ACTION Note of ITS 3.3.4, ITS 3.3.5, ITS 3.3.6, and ITS 3.3.7 allows separate condition entry for each parameter. This Note is needed in the ITS since the ACTIONS are not prescribed separately. This is a change in the presentation of requirements, with no actual change in requirements and is, therefore, administrative. The change is consistent with the NUREG.

A7 Not used.

A8 CTS Table 3.5.1-1, Column (D) and related Note (e) require the reactor to be in hot shutdown (equivalent to ITS MODE 3) in 12 hours and in Cold Shutdown (equivalent to ITS MODE 5) within 24 hours if the channel is not restored within 48 hours after hot shutdown. ITS 3.3.5 Required Action B.2.1 and B.2.2 is to reduce reactor coolant system pressure below the applicability, when one or more channels are inoperable or Required Action A.1 and associated Completion Time cannot be met. CTS 3.5.3 Notes (1) and (2) allow the reactor coolant system pressure parameter trip functions to be bypassed below 1750 and 900 psig respectively. Considering the allowance to bypass, the CTS action is essentially the same as the ITS ACTION. As such, the proposed change is considered administrative and consistent with the NUREG.

A9 CTS Table 4.1-1 Surveillance frequencies are replaced with those from the NUREG. The CTS and corresponding ITS Frequencies are as follows:

CTS	ITS
ES - Each shift	12 hours
DA - Daily	24 hours
WE - Weekly	7 days
MO - Monthly	31 days
QU - Quarterly	92 days
AN - Annually	12 months
PS - Prior to startup if not performed previous week	Not Used
NA - Not Applicable	Not Used
2 STB- STAGGERED TEST BASIS	STAGGERED TEST BASIS

Each of these changes is consistent with the current application of the CTS frequencies at ONS. These changes maintain requirements consistent

This change represents no actual change in requirements, only a change in presentation of requirements. This change is consistent with the NUREG.

A14 CTS 3.3.4 requires the BWST level instrumentation to be OPERABLE when the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical. ITS 3.3.8 requires this instrumentation to be OPERABLE during MODES 1, 2, and 3. MODE 3 is defined as subcritical with the average coolant temperature > 250°F. CTS criteria specified as 250°F is considered more limiting than the 350 psig criteria, since the saturation temperature of water at 350 psig is > 435°F. As such, the proposed change is considered administrative. In addition, the CTS applicability statement "with fuel in the core" is deleted since the ITS definition of MODE is premised on "fuel in the vessel." This is a format change due to ITS conversion and is administrative in nature. The proposed change is consistent with the NUREG.

A15 The CTS Table 4.1-1 Item 62 specifies the testing requirements for the Main Steam Header Pressure and MSLB detection (analog) channels. These testing requirements are retained in ITS SR 3.3.11.2. As applied at ONS, CTS Table 4.1-1 requires a CHANNEL CALIBRATION and CHANNEL 2 FUNCTIONAL TEST of this function every 18 months. Both the CTS and ITS definitions specify that the required calibration includes the CHANNEL FUNCTIONAL TEST. Therefore, the specific requirement to perform the CHANNEL FUNCTIONAL TEST is not retained in the ITS. This change represents no actual change in requirements, only a change in presentation of requirements and is consistent with the NUREG.

A16 CTS 3.5.7 Applicability requires the Main Steam Line Break (MSLB) detection and feedwater isolation circuitry to be OPERABLE when main steam header pressure is greater than 700 psig. ITS 3.3.11, 12, and 13 Applicability for this circuitry of MODES 1, 2, and MODE 3 when main steam header pressure is greater than 700 psig except when all MFCVs and SFCVs are closed. With the exception of "except when all MFCVs and SFCVs are closed," which is addressed in a separate less restrictive DOC, the CTS and ITS Applicabilities are equivalent. Therefore, the change is administrative and is consistent with the NUREG.

A17 CTS 3.5.6, 3.4.1, 3.4.3, and 3.3.4 provide requirements for Post Accident Monitoring (PAM) Functions. ITS 3.3.8 consolidates the CTS PAM Functions into one Specification. In CTS, each Function has separate Actions and is considered separately. However, since ITS addresses the PAM functions using common actions, Note 2 to the ACTIONS Table is used to indicate separate Condition entry is permitted for each function. This is a change in the presentation of requirements, with no actual change in requirements and is, therefore, administrative. The change is consistent with the NUREG.

A18 CTS 3.5.6.2 in conjunction with Table 3.5.6-1, Column B serves as a pointer to the appropriate action for each function (ITS PAM Functions

A35 CTS Table 3.5.1-1, Item 14 describes the functional unit as ESF Reactor Building Isolation (Essential Systems) and Reactor Building Cooling System. ITS LCO 3.3.6.c describes this function as Reactor Building (RB) Cooling, Reactor Building Essential Isolation and Penetration Room Ventilation (ES Channels 5 and 6). The ITS LCO statement was expanded to provide a complete list of equipment actuated by the ES channels. The LCO statement does not change the technical specification requirements. Therefore, the change is administrative.

A36 CTS 3.8.1.10 requires the reactor building purge system, including the radiation monitors, to be operable immediately prior to refueling operation. ITS 3.3.16 requires the Reactor Building Isolation-High Radiation function to be OPERABLE during CORE ALTERATION and during movement of irradiated fuel assemblies within the reactor building. CTS defines "refueling operation" as an operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed. Also, movement of irradiated fuel assemblies within the reactor building can only be performed subsequent to the start of refueling. As such, the change is administrative and consistent with the NUREG.

2 A37 CTS Table 4.1-1 Remarks Note 1 allows a one-time extension of the test
2 frequency to a maximum of 23 months for items 45 and 46 and a maximum
2 of 24 months for items 58, 59, 60, and 61 for Oconee Unit 2 in
2 operating cycle 16. This provision is no longer needed and is deleted
2 since operating cycle 16 has passed. As such, the change is
2 administrative.

ITS Section 3.3

ID 118

Subject: Corrected Section 3.3 DOC M33 to refer to 31 day Frequency.

M29 CTS does not require a calibration of the Wide Range Nuclear instrumentation. ITS SR 3.3.8.3 requires a complete check of the instrument loop and sensor. The addition of this CHANNEL CALIBRATION is appropriate since it verifies the channel responds to measured parameters within the necessary range and accuracy. The proposed change is more restrictive since it is an additional restriction on operation and is consistent with the NUREG.

M30 CTS Table 3.5.1-1 essentially does not have any required actions for ITS 3.3.1 Functions 1.b and 11 when in MODE 3 or lower since Column D only requires the unit be placed in hot shutdown (equivalent to ITS MODE 3). Consistent with the proposed ITS applicability (Refer to DOC M2), required actions are provided that require the unit be placed in a condition outside the applicability of the LCO when the Required Actions cannot be met. Required Action D.1 is added to require the operator to open all the CRD trip breakers in 6 hours. This additional restriction on operation is appropriate since it ensures rod motion is not possible when the required trip channels are inoperable. This change is consistent with the NUREG.

M31 Not used.

M32 Not used.

M33 CTS Table 4.1-1, Column "Check" for Items 54, 57, 58, and 59, currently does not require a check of the containment high range radiation monitor, containment hydrogen monitor, wide range hot leg level, and reactor vessel head level instrument channels. ITS SR 3.3.8.1 is added
2 to require a CHANNEL CHECK of these instrument channels every 31 days consistent with the NUREG. A CHANNEL CHECK provides reasonable assurance that a gross failure of instrumentation will be identified promptly. The proposed change is more restrictive since it is an additional restriction on operation and is consistent with the NUREG.

M34 CTS 3.7.4 LCO Applicability for Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits is above Cold Shutdown. ITS 3.3.18 LCO Applicability for Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits is MODES 1, 2, 3, 4, 5, and 6 and during movement of irradiated fuel assemblies. ITS MODES 1, 2, 3, and 4 are considered equivalent to the CTS applicability of above Cold Shutdown. ONS design requires that the voltage sensing circuit associated with an AC power source be OPERABLE for the AC power source to be considered OPERABLE. Therefore, since requirements for AC Source in MODES 5 and 6 and during movement of irradiated fuel assemblies are added (refer to Section 3.8), requirements for EPSL voltage sensing circuits must be added. These additional applicability requirements provide assurance that systems are available to provide adequate coolant inventory makeup, to mitigate a fuel handling accident,and to mitigate the effects of events that can lead to core damage during shutdown and that instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition. LCO Note 2 is added to

specify that only the EPSL voltage sensing circuit(s) associated with required AC power source(s) are required to be OPERABLE. ITS 3.3.18 ACTION C is added to require the affected AC Source to be declared inoperable when the Required Action and associated Completion Time is not met or when two or more channels of a required circuit(s) are inoperable in MODES 5 and 6. ITS 3.3.18 ACTION D is added to require suspending movement of irradiated fuel assemblies when the Required Action and associated Completion Time is not met during movement of irradiated fuel assemblies. Since there are currently no EPSL requirements during cold shutdown and refueling shutdown, the proposed change is more restrictive since it is an additional restriction on operation.

M35 CTS Table 4.1-1 provides no specific CHANNEL CALIBRATION requirement for the Nuclear Overpower High and Low Setpoints. The High and Low Setpoints are calibrated administratively during reactor shutdowns and reactor startups. ITS SR 3.3.1.5is added for comparable ITS Functions 1.a and 1.b to provide an explicit 18 month CHANNEL CALIBRATION for these functions. This test ensures the channel responds to the measured parameter within the necessary range and accuracy and leaves the channels adjusted to account for instrument drift to ensure that the instrument channel will remain operational between successive tests. A Note to the SR specifically excludes neutron detectors from this CHANNEL CALIBRATION. The addition of SR 3.3.1.5is considered an appropriate restriction on unit operation since the accident analyses takes credit for these reactor trip functions. The addition of this requirement represents a more restrictive change and is consistent with the NUREG.

M36 CTS 3.4.1 does not allow the reactor to be heated above 250°F unless each emergency feedwater flow (EFW) path has at least one flow indicator operable (CTS 3.4.1.b). ITS 3.3.8, Item 21, Emergency Feedwater Flow, is added to require two channels of EFW flow to be operable in MODES 1, 2 and 3, along with associated Required Actions A, B, and G, the Table entry, and Notes that are applicable for this PAM function. This is consistent with the ONS Regulatory Guide 1.97 Safety Evaluation Report which identifies these indicators as a Category 1 variable and is appropriate to ensure its availability post accident since EFW flow is the primary indication used by the operator to verify that the EFW System is delivering the correct flow to each steam generator. ITS 3.3.8 ACTION A limits the time that one of the two channels can be inoperable to 30 days. If the channel is not restored within 30 days then ITS 3.3.8 ACTION B requires that a written report be submitted to the NRC which identifies the proposed restorative actions and discusses the root cause evaluation. ITS 3.3.8 ACTION G is a pointer to further appropriate action (dependent upon the PAM function) when the Required Action and associated Completion Times for inoperable channels are not met. The addition of these requirements represents a more restrictive change and is consistent with the NUREG.

M37 CTS 3.4.3.b requires the unit be in hot shutdown (equivalent to ITS MODE 3) within 12 hours when no EFW flow indicators in a flow path are

operable and below 250°F (equivalent to ITS MODE 4) within another 12 hours. ITS 3.3.8 Required Action H.2 requires the unit be placed in MODE 4 within 6 hours after reaching MODE 3. The ITS Completion time for Required Action H.2 is more restrictive since the unit must be placed in MODE 4 six hours earlier. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging plant systems. The addition of these requirements represents a more restrictive change and is consistent with the NUREG.

M38 CTS does not require the Manual Keowee Emergency Start function to be OPERABLE below Cold Shutdown. ITS Specification 3.3.22, including appropriate LCO, ACTIONS, and SRs, is added to require one channel of the Manual Keowee Emergency Start function to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies. This is necessitated by the addition of requirements for AC Source in MODES 5 and 6 and during movement of irradiated fuel assemblies (refer to Section 3.8). Required Action A.1 requires both Keowee Hydro Units to be declared inoperable immediately when the required channel is inoperable. ITS SR 3.3.22.1 requires a CHANNEL FUNCTIONAL TEST of the Keowee manual emergency start function every 12 months. This function is currently tested during the performance of CTS 3.7.1.11. CTS 3.7.1.11 uses the manual start function as a method of initiating the Keowee Hydro Units when verifying they start within 23 seconds and synchronize to the grid. These additional requirements provide assurance that the required AC Sources are available during shutdown. This ensures that systems are available to provide adequate coolant inventory makeup, to mitigate a fuel handling accident, and to mitigate the effects of events that can lead to core damage during shutdown and that instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition. Since there are currently no EPSL requirements during cold shutdown and refueling shutdown, the proposed change is more restrictive since it is an additional restriction on operation.

M39 CTS Table 3.5.1-1 Note (c) allows continued operation above hot shutdown with the required source range instrument channel inoperable when 2 of 4 wide range instrument channels are indicating greater than 4E-4% rated power. ITS 3.3.9 ACTION C, which continues to allow operation above MODE 3 with the required source range instrument channel inoperable, also requires action be initiated to repair the inoperable instrument channel within 1 hour. This more restrictive action is appropriate to ensure future availability. The 1 hour Completion Time is sufficient to initiate the action. This change is consistent with the NUREG.

M40 CTS 3.5.1.1 Applicability for the TSV Closure instrumentation channels is while in the startup mode or when the reactor is in a critical state. ITS 3.3.15 Applicability for the TSV Closure instrumentation channels is in MODES 1, 2, and 3 except when all TSVs are closed. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq$ 0.99 k_{eff}. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The expanded applicability is appropriate since during MODE 3 conditions there is significant mass and energy in the RCS and steam generators and the TSV Closure function is needed to stop steam flow to the turbine (to prevent overcooling) following a reactor trip. As such, the addition of applicability in MODE 3 is more restrictive. The more restrictive requirement is consistent with comparable NUREG (3.3.11-1, Function 4) requirements.

ITS Section 3.3

ID 121

Subject: Revised Section 3.3, Attachment 5, JFD 8 to replace "TSTF-xxx" with "TSTF-264."

7 NUREG SR 3.3.1.7, SR 3.3.5.4 and SR 3.3.11.4 are not adopted. Consistent with current licensing basis, response time testing of RPS, ESPS, and EFIC (ONS equivalent) circuitry is not performed. Plant equipment does not readily lend itself to such testing.

2 8 TSTF-264 clarifies that agreement criteria include one decade of overlap during a power increase when transitioning between the intermediate range monitors and the power range monitors. ONS uses wide range monitors which have an upper range up to 200% power, therefore, there is no need to transition between monitors. As such, this portion of TSTF is not included.

9 The unit specific design of the ONS ESPS provides for three analog channels of instrumentation for each of the monitored parameters. These three channels provide the required input to each of the eight automatic actuation logic channels. Contrary to the system design depicted in the requirements of the NUREG, these three instrument channels provide input to both trains of automatic actuation logic channels. This unit specific design difference required the deletion of the phrase "in each ESFAS train" from NUREG LCO 3.3.5 as well as appropriate changes to the Bases.

10 Not used.

11 NUREG Table 3.3.5-1 lists equipment actuated by each of the ESPS Parameters. This list is incomplete and has not been included in the ITS in favor of the more complete list provided in the Bases. Also, the term "Setpoint" is removed from the Parameter title in the Table and from Required Actions B.2.1, B.2.2 and B.2.3 since the setpoint is not a parameter. Referring to a setpoint as a parameter is inconsistent with the identification of other parameters and functions throughout the NUREG. Removal of this information represents no actual change in requirements.

12 In the conversion to ITS, NUREG Table 3.3.17-1, Post Accident Monitoring Instrumentation, is modified by Note c which indicates that the containment isolation valve position indication requirements apply only to containment isolation valves that are electrically controlled. This is consistent with ONS Regulatory Guide 1.97 response for CIV position indication and the NRC's Safety Evaluation Report for this response.

13 Not used.

14 The Functions specified in NUREG LCO 3.3.6 are modified to match the Functions as presented in the CTS, UFSAR, and other design basis documents. NUREG Specification 3.3.7 has also been modified to include ONS unit specific terminology. These changes were made to provide requirements consistent with the design of the ONS and consistent with the specific terminology and names associated with the ONS ESPS.

ITS Section 3.3

ID 122

Subject: Revised Section 3.3, Attachment 3, DOC L8 to delete reference to SR 3.3.1.4 which was deleted as part of TSTF 259.

allow continued reactor operation for an indefinite period when one of three ESPS channels is inoperable provided the inoperable channel is placed in a tripped condition for actuation within one hour. This less restrictive provision is acceptable since this action leaves the system in a one-out-of-two condition for actuation. Thus, if another channel fails, the ESPS instrumentation can still perform its actuation functions. This less restrictive change is consistent with the NUREG.

L6 CTS does not include an allowance that allows delay of entry into actions when a channel is made inoperable for testing. The Note for ITS SR 3.3.5.2 allows a delay of up to 8 hours in the entry into the associated Condition and Required Action for the performance of this CHANNEL FUNCTIONAL TEST provided the remaining two instrumentation channels are OPERABLE or tripped. This Note provides a relaxation of ACTION requirements which is less restrictive than the application of CTS requirements. This Note provides a reasonable amount of time to perform the required testing while still allowing the channel being tested to remain in an untripped state. Additionally, the design of the ESPS system will not allow complete testing of an instrument channel with the channel in a tripped state. Therefore, placing the channel in a tripped state, as required by ITS 3.3.5 Required Action A.1, prevents the completion of the testing required by SR 3.3.5.2. This change is consistent with the NUREG.

L7 CTS Table 3.5.1-1 requires a unit shutdown in 12 hours when one or more ESPS channels in one or more functions are inoperable. ITS 3.3.6 Required Action A.1 and its associated Completion Time provide a 72 hour time period in which the unit may continue operation, with one or more ESPS Functions having one channel of the manual initiation feature inoperable, prior to entering an ACTION which results in the unit entering MODE 3. This change is made to provide ACTION requirements consistent with the safety function of the system, considering the allowed outage time for the actuated system. Therefore the less restrictive change is considered appropriate. This change is consistent with the NUREG.

L8 CTS Table 4.1-1 Item 4 requires a calibration of the power range instruments against the incore instruments monthly. This calibration is also required to be performed within some unspecified period of time after each startup if not performed within the previous week. These CTS
2 requirements are replaced by ITS SR 3.3.1.3. The ITS SR with its
2 specified 31 day Frequency represents less restrictive requirements in that the calibration is no longer required following each startup if not performed within the previous week. Removal of the required calibration following each startup, is acceptable because deviation between the AXIAL POWER IMBALANCE indicated by the power range instruments and that indicated by the incore instruments generally occurs slowly. The 31 day Frequency is consistent with NUREG.

LESS RESTRICTIVE CHANGE L8

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS Table 4.1-1 Item 4 requires a calibration of the power range instruments against the incore instruments monthly. This calibration is also required to be performed within some unspecified period of time after each startup if not performed within the previous week. These CTS requirements are replaced by
2 ITS SR 3.3.1.3. The ITS SR with its specified 31 day Frequency represents less restrictive requirements in that the calibration is no longer required following each startup if not performed within the previous week. Removal of the required calibration following each startup, is acceptable because deviation between the AXIAL POWER IMBALANCE indicated by the power range instruments and that indicated by the incore instruments generally occurs slowly. The 31 day Frequency is consistent with NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change does not result in any hardware changes or changes in operating methods. The change removes an unnecessary additional performance of a surveillance which has been performed within its normal monthly Frequency. Not performing the surveillance at the startup would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

ITS Section 3.3

ID 126

Subject: Revised Section 3.3 DOC M2 to reference TSTF 218 in place of JFD 24.

RODS are withdrawn or capable of withdrawal. The automatic insertion of any withdrawn CONTROL ROD is consistent with evaluations of accidents initiated from MODE 3.

CTS requires the Nuclear Overpower - High Setpoint and RCS High Pressure function to be OPERABLE when the reactor is in a startup mode or in a critical state (equivalent to ITS MODES 1 and 2 as discussed above). ITS 3.3.1 requires these functions to be OPERABLE during MODES 1 and 2 and MODE 3 when not in shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal (ITS Table 3.3.1-1, Note d). The additional requirement for OPERABILITY in MODE 3 is considered appropriate to ensure that the instrumentation required to initiate the insertion of any withdrawn CONTROL RODS is OPERABLE whenever CONTROL RODS are withdrawn or capable of withdrawal. The automatic insertion of any withdrawn CONTROL ROD is consistent with evaluations of accidents initiated from MODE 3.
The proposed more restrictive changes to the Applicability for RPS functions are appropriate since they provide additional assurance that a reactor trip will be actuated if needed. These changes are
2 consistent with the NUREG as modified by TSTF-218.

M3 CTS 3.5.1.1 requires the RPS manual trip function, Reactor Trip Module, and CRD trip devices of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. The ITS applicability for the manual reactor trip function (ITS 3.3.2), the RTMs (ITS 3.3.3) and the CRD trip devices (ITS 3.3.4) is MODES 1 and 2 and MODES 3, 4, and 5 during shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq$ 0.99 k_{eff}. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. Since the ITS is applicable under more conditions than the current applicability, the ITS Applicability is more restrictive. The proposed applicability provides additional assurance that a reactor trip will be actuated if needed to prevent accident conditions from exceeding those calculated in the accident analyses. This change is consistent with the NUREG.

CTS Table 3.5.1-1 does not have any required actions for the manual reactor trip function, the RTMs and the CRD trip devices when in MODE 3 or lower since Column D only requires the unit be place in hot shutdown (equivalent to ITS MODE 3). Consistent with the proposed ITS applicability, required actions are provided for each ITS Specification that require the unit be placed in a condition outside the applicability of the Specification when the LCO Required Actions cannot be met. ITS 3.3.2 Required Action C.1 is added for the manual reactor trip function to require the operator to open all the CRD trip breakers in 6 hours. Appropriate Required Actions are provided for the RTM (ITS 3.3.3 Required Action C.1 and C.2) and CRD trip devices (ITS 3.3.4

ITS Section 3.3

ID 131

Subject: NUREG page B 3.3-25 - TSTF 264 is marked out and annotated with JFD 8. Revised to correctly annotate with JFD 5.

BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION ~~and RPS RESPONSE TIME~~ testing.

The SRs are modified by a Note. The ~~first~~ Note directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

Reviewer's Note: The CHANNEL FUNCTIONAL TEST Frequencies are based on approved topical reports. For a licensee to use these times, the licensee must justify the Frequencies as required by the NRC Staff SER for the topical report.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally 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 two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined ~~by the unit staff~~ based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency, ~~about once~~ equivalent to every shift, is based on operating experience that demonstrates channel failure is rare. Since

Supp 2

Supp. 1

INSERT TSTF 264

(continued)

ITS Section 3.3

ID 135

Subject: Revised Section 3.3 DOC M25 to remove discussion of 246 F and only discuss in terms of "when SG relied upon for heat removal.

APPLICABILITY for EFW pump initiation circuitry is MODES 1, 2, & 3 and MODE 4 when a steam generator is relied upon for heat removal.

2 Requiring the EFW pump initiation circuitry to be OPERABLE in MODE 4
2 when relied upon for decay heat removal is a more restrictive requirement upon unit operation and is consistent with ITS 3.7.5 for the EFW System and the corresponding NUREG Specification.

M26 CTS does not include all Type A and Category 1 post accident monitoring (PAM) instrumentation identified in the plant specific Regulatory Guide 1.97 response and associated NRC Safety Evaluations. ITS 3.3.8 incorporates all Type A and Category 1 PAM Functions consistent with the NUREG. The following PAM Functions, including the associated LCO, Applicability, ACTIONS, Table entries, and Notes, are added:

2. RCS Hot Leg Temperature
4. RCS Pressure (Wide Range)
8. Containment Isolation Valve Position
11. Pressurizer Level
12. Steam Generator Water Level
13. Steam Generator Pressure
15. Upper Surge Tank Level
18. HPI System Flow
19. LPI System Flow
20. Reactor Building Spray Flow

Surveillance Requirements are added (ITS SRs 3.3.8.1 and 3.3.8.3) for PAM Functions 2, 4, 8, 13, 15, and 20. SR 3.3.8.1 is added for PAM Functions 18 and 19. CTS provides comparable Surveillance requirements for pressurizer level (PAM 11)and steam generator water level (PAM 12) indicators (CTS Table 4.1-1, items 26 and 39 respectively). The addition of SR 3.3.8.1 for the Functions designated above is appropriate since a CHANNEL CHECK provides assurance that gross channel failure will be detected and is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The addition of SR 3.3.8.3 for the Functions designated above is appropriate since the CHANNEL CALIBRATION verifies the channel responds to measured parameters within the necessary range and accuracy.

Type A variables are included in the ITS because they provide the primary information that permits the control room operator to take specific manually controlled actions that are required when no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). Additionally, Category 1 variables are the key variables deemed risk significant because they are needed to: a) determine whether systems important to safety are performing their intended functions; b) provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and c) provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any

impending threat. Since these PAM Functions are not in the CTS requirements, their addition represents a more restrictive change.

M27 The CTS applicability for Table 3.5.6-1 instrument 8 is "when RCS temperature is > 300°F". The CTS applicability for other Table 3.5.6-1 PAM instruments is "above hot shutdown." ITS 3.3.8 Applicability for PAM functions is MODES 1, 2, and 3. The CTS applicability of "above hot shutdown" is equivalent to ITS MODES 1 and 2. The CTS applicability of "when RCS temperature is > 300°F" is equivalent to ITS MODES 1 and 2 and part of MODE 3 (Note: transition to MODE 4 is at 250°F). Therefore, the ITS applicability is more restrictive since OPERABILITY is required in MODE 3, but appropriate since these variables are related to the diagnosis and actions required to mitigate accidents that are assumed to occur in these MODES. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low, therefore, PAM instrumentation is not required to be OPERABLE in these MODES. The proposed change is consistent with the NUREG.

Consistent with the more restrictive ITS applicability, Required Action H.2 is added for CTS Table 3.5.6-1 instruments 1 through 7 and 9 requiring the unit be placed in MODE 4 within 18 hours. This is appropriate since it removes the unit from the applicability of the LCO. For CTS Table 3.5.6-1 Instrument 8 (ITS Table 3.3.8-1, Function 17), Action 4 requires the unit to be in hot shutdown within the next 12 hours and below 300°F within the next 24 hours. For this Function, ITS Required Action H.2 requires the unit be in MODE 4 (below 250°F) in 18 hours from the time of entry into the condition versus the total of 36 hours currently allowed. This change is consistent with the NUREG.

M28 CTS SR 3.7.5.1 requires performance of SR 3.7.1.14.1 (EPSL automatic transfer) on a Frequency specified in the applicable SR. CTS SR 3.7.3.1 requires performance of SR 3.7.1.14 on a Frequency specified in the applicable SR. ITS SR 3.3.17.1 and SR 3.3.21.1 require a CHANNEL FUNCTIONAL TEST of the automatic transfer function channels and the Keowee automatic start channels. The CTS test requirement (CTS SR 3.7.1.14) performs a functional verification for the source and Main Feeder Bus voltage sensing, Keowee Emergency start, Loadshed and Transfer-to-Standby, and Retransfer-to-Startup logic of the EPSL System by performing an automatic transfer of the Main Feeder Buses to the Startup Transformer, Standby Buses and retransfer to the Startup Transformers. The method of performing this test is relocated to the Bases for SR 3.3.17.1 and 3.3.21.1 as discussed in DOC LA12. These CTS SRs are captured by the CHANNEL FUNCTIONAL TEST requirements of ITS SR 3.3.17.1 and SR 3.3.21.1. However, the CHANNEL FUNCTIONAL TEST requirements are considered more prescriptive in that they require the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display, and trip functions. As such, the proposed change is more restrictive on plant operation and consistent with the NUREG.

ITS Section 3.3
3.7

ID 142

Subject: Incorporated latest revision to TSC 95-03 - MSLB

Specification 3.3.11

(A1) (except as marked)

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

except when all MFCVs and SFCVs are closed

Applicability (MODES 1 + 2 and MODE 3) (A16)

(L18)

Applic Applies to main steam line break (MSLB) detection and feedwater isolation circuitry when main steam header pressure is greater than 700 psig and to the Main Feedwater main and startup control (Main Feedwater control) valves when Reactor Coolant System temperature is greater than 250 °F. (SEE 3.7)

Objective

To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside containment.

Specifications

LCO 3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items 20, 21, and 22. (SEE 3.3.12+13)

3.5.7.2 The Main Feedwater control valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

a. A Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed within 8 hours from discovery and verified closed once per 7 days.

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a hot shutdown condition within 12 hours, and be less than or equal to an RCS temperature of 250 °F in an additional 18 hours.

(SEE 3.7)

Bases (A2)

Suppl 2 The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner. 1

Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main

Specification 3.3.12

A1 (except as marked)

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

except when all MFCVs and SFCVs are closed

Applicability

MODES 1 + 2 and MODE 3 — A16

Applic

Applies to main steam line break (MSLB) detection and feedwater isolation circuitry when main steam header pressure is greater than 700 psig and to the Main Feedwater main and startup control (Main Feedwater control) valves when Reactor Coolant System temperature is greater than 250 °F.

L18

<SEE 3.7>

Objective

To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside containment.

Specifications

LCO

3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items 20, 21, and 22.

<SEE 3.3.11 + 13>

3.5.7.2 The Main Feedwater control valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

a. A Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed within 8 hours from discovery and verified closed once per 7 days.

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a hot shutdown condition within 12 hours, and be less than or equal to an RCS temperature of 250 °F in an additional 18 hours.

<SEE 3.7>

Bases

A2

The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner.

Suppl 2

Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main

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Specification 3.3.13

(A1) <except as marked>

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

Applicability (MODES 1 + 2 and MODE 3) (A16)

except when all MFCVs and SFCVs are closed (L18)

Applic Applies to main steam line break (MSLB) detection and feedwater isolation circuitry when main steam header pressure is greater than 700 psig and to the Main Feedwater main and startup control (Main Feedwater control) valves when Reactor Coolant System temperature is greater than 250 °F. <SEE 3.7>

Objective

To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside containment.

Specifications

LCO 3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items (20, 21, and 22) <SEE 3.3.11 + 12>

3.5.7.2 The Main Feedwater control valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

a. A Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed within 8 hours from discovery and verified closed once per 7 days.

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a hot shutdown condition within 12 hours, and be less than or equal to an RCS temperature of 250 °F in an additional 18 hours.

<SEE 3.7>

Bases (A2)

The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner. Suppl 2 |

Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main

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ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Specification 3.4.10

(A1) (except as marked)

(A13) in accordance with the Inservice Testing Program

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

SR 3.4.10.1

	Item	Test	Frequency	
1.	Control Rod Movement [1]	Movement of Each Rod	Monthly	(SEE 3.1)
2.	Pressurizer Safety Valves	Setpoint	~~18 months~~ [4]	
3.	Main Steam Safety Valves	Setpoint	18 months [4]	(SEE 3.7)
4.	Refueling System Interlocks [5]	Functional	Prior to Refueling	(SEE 3.3)
5.	Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly	(SEE 3.7)
6.	Reactor Coolant System Leakage [2]	Evaluate	Daily	(SEE 3.4.13)
7.	Emergency Condenser Circulating Water System Test [6]	Functional	18 months	
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	(SEE 3.7)
9.	Spent Fuel Cooling System	Functional	Prior to Refueling	
10.	High Pressure and Low Pressure Injection System [3]	Vent Pump Casings	Monthly and Prior to Testing	(SEE 3.5)
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	(SEE 3.7)
12.	MSLB Feedwater Isolation Feature [7]	Functional	18 months	
13.	Essential Siphon Vacuum System Test [8]	Functional	Quarterly	

Supp 2

Supp 2

Supp 2

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.4.10

(A1) <except as marked>

(1) Applicable only when the reactor is critical. <SEE 3.1 + 3.7>

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. <SEE 3.4.13>

(3) Operating pumps excluded. — <SEE 3.5>

SR 3.4.10.1 Supp 2

(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. (A13) <SEE 3.3>

(5) Applicable only to the interlocks associated with the Reactor Building Purge System.

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System. <SEE 3.7>

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature.

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit.

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.4.12

(A1) (except as marked)

4.2.6 The power operated relief valve (PORV) is used for low temperature overpressure protection of the RCS and shall be demonstrated operable by:

Suff. 2

a. ~~Performing an operability test prior to each startup from cold shutdown.~~ — LA1

SR 3.4.12.7

b. Performing a calibration of the actuation circuit every 18 months.

c. ~~Performing an inspection of the PORV at least once every two refueling cycles.~~ — LA5

4.2.7 ~~Each shift, the RCS vent(s) (as defined in Specification 3.1.2.9) shall be verified to be open, if the vent(s) is(are) being used for overpressure protection. If the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then these valves will open at least once per 31 days.~~ — LA9

Bases

The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10 CFR 50.55(a) to the extent practicable within limitations of design, geometry and materials of construction. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties. — A2

Add SR 3.4.12.1
SR 3.4.12.2
SR 3.4.12.3
SR 3.4.12.4
SR 3.4.12.5
SR 3.4.12.6

M30

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Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Specification 3.4.13

(A) (except as marked)

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency	
1. Control Rod Movement [1]	Movement of Each Rod	Monthly	(SEE 3.1)
Supp 2 | 2. Pressurizer Safety Valves	Setpoint	18 months[4]	(SEE 3.4.10)
Supp 2 | 3. Main Steam Safety Valves	Setpoint	18 months[4]	(SEE 3.7)
Supp 2 | 4. Refueling System Interlocks[5]	Functional	Prior to Refueling	(SEE 3.3)
5. Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly	(SEE 3.7)
Supp 2 | SR 3.4.13.1 6. Reactor Coolant System [2] Leakage	Evaluate (M35) Verify RCS operational leakage is within limits	Daily (12 hrs) (L14)	
7. Emergency Condenser [6] Circulating Water System Test	Functional	18 months	
8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	(SEE 3.7)
9. Spent Fuel Cooling System	Functional	Prior to Refueling	
10. High Pressure and Low [3] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	(SEE 3.5)
Supp 2 | 11. Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	(SEE 3.7)
12. MSLB Feedwater Isolation [7] Feature	Functional	18 months	
13. Essential Siphon Vacuum [8] System Test	Functional	Quarterly	

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.4.13

(A1) <except as marked>

Applic ↓ SR 3.4.13.1 Note

(1) Applicable only when the reactor is critical. <SEE 3.1 + 3.7>

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. MODES 1, 2, 3 & 4 operation

(3) Operating pumps excluded. —<SEE 3.5>

Supp 2

(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. <SEE 3.4 + 3.7>

(5) Applicable only to the interlocks associated with the Reactor Building Purge System. <SEE 3.3>

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature.

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit. <SEE 3.7>

Specification 3.4.14

Add Note to SR 3.4.14.1 — M39

Supp. 2

4.5.1.1.3 Core Flooding System

a. Every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During pressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests — See 3.5

4.5.1.2.1 Valves - Power Operated

Supp. 2

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

b. Every 18 months, the following LPI system valves shall be cycled manually to verify the manual operability of these power operated valves:

(1) LPI pump discharge (ES) LP-17,-18
(2) LPI discharge throttling LP-12,-14
(3) LPI discharge header crossover LP-9,-10
(4) LPI discharge to HPI/RBS LP-15,-16 — M39

M40

4.5.1.2.2 Check Valves

PIVs — 7 days — L3

entering MODE 2 — SR 3.4.14.1

Periodic individual leakage testing[a] of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakage rates and limiting conditions for operation, see Technical Specification 3.1.6.10. — LA13

Bases

The Emergency Core Cooling Systems are the principle reactor safety features in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. The HPI system test required by Specification 4.5.1.1.1 verifies that the HPI system responds as required to actuation of ES channels 1 and 2.

A2

LA7

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

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Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

ITS Section 3.4

ID 82

Subject: Revised 3.4.12 to reinstate NUREG terminology of "isolate affected CFT" versus "deactivate affected CFT."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with high pressure injection (HPI) deactivated, and the core flood tanks (CFTs) 2 isolated and:

a. An OPERABLE power operated relief valve (PORV) with a lift setpoint of $\leq$ 480 psig; and

b. Administrative controls implemented that assure $\geq$ 10 minutes are available for operator action to mitigate an LTOP event.

APPLICABILITY: MODE 3 when any RCS cold leg temperature is $\leq$ 325°F,
MODES 4, 5, and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open.

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

2 1. CFT isolation is only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in Specification 3.4.3.

2. The PORV is not required to be OPERABLE when no HPI pumps are running and RCS pressure < 100 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. HPI activated.	A.1 Initiate action to deactivate HPI.	Immediately
2 B. A CFT not isolated when CFT pressure is greater than or equal to the maximum RCS pressure for existing temperature allowed by Specification 3.4.3.	B.1 Isolate affected CFT.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 4 with RCS temperature > 233°F. OR C.2 Depressurize affected CFT to < 373 psig.	12 hours 12 hours
D. PORV inoperable.	D.1 Restore PORV to OPERABLE status.	1 hour
E. Required Action and associated Completion Time of Condition D not met.	E.1 Be in MODE 3 with RCS average temperature > 325°F. OR E.2 Depressurize RCS to < 100 psig.	23 hours 35 hours

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
	SR 3.4.12.1	Verify HPI is deactivated.	12 hours
2	SR 3.4.12.2	Verify each CFT is isolated.	12 hours
	SR 3.4.12.3	Verify pressurizer level is ≤ level necessary to assure ≥ 10 minutes are available for operator action to mitigate an LTOP event.	30 minutes during RCS heatup and cooldown AND 12 hours
	SR 3.4.12.4	Verify PORV block valve is open.	12 hours
	SR 3.4.12.5	Perform CHANNEL FUNCTIONAL TEST for PORV.	Within 12 hours after decreasing RCS temperature to ≤ 325°F AND 31 days thereafter

(continued)

BASES

BACKGROUND (continued)

The LTOP approach to protecting the vessel by limiting coolant addition capability requires administrative controls upon RCS makeup flow, and requires deactivating HPI, and isolating the core flood tanks (CFTs). 2 2

Should an HPI pump inject on an HPI actuation, the pressurizer level and PORV may not prevent overpressurizing the RCS.

The administrative controls upon pressurizer level provides a compressible vapor space or cushion (either steam or nitrogen) that can accommodate a coolant insurge and prevent a rapid pressure increase, allowing the operator time to stop the increase. The PORV, with reduced lift setting, is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI deactivated, the ability to provide RCS coolant addition is restricted. To balance the possible need for coolant addition, the LCO does not require the makeup system to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the makeup system can provide flow with the HPI pumps providing RCS makeup through the makeup control valve.

PORV Requirements

As required for the LTOP, the PORV is signaled to open if the RCS pressure approaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the PORV is signaled to open. Maintaining the setpoint within the limits of the LCO ensures the Reference 1 limits will be met in any event analyzed for LTOP.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases steam, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

due to neutron embrittlement. Each time the P/T limit curves are revised, the LTOP System will be re-evaluated to ensure that its functional requirements can still be met with the PORV and pressurizer level/administrative controls method.

Transients that are capable of overpressurizing the RCS have been identified and evaluated. These transients relate to either mass input or heat input: actuating the HPI System, discharging the CFTs, energizing the pressurizer heaters, failing the makeup control valve open, losing decay heat removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and adding nitrogen to the pressurizer.

HPI actuation and CFT discharge are the transients that may result in exceeding P/T limits within < 10 minutes in which time no operator action is assumed to take place. Starting an RCP and adding nitrogen to the pressurizer are self limiting events. In the rest, operator action after that time precludes overpressurization. The analyses demonstrate that the time allowed for operator action is adequate, or the events are self limiting and do not exceed P/T limits.

The following are required during the LTOP MODES to ensure that transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Limiting RCS makeup flow capability;

b. Deactivating HPI; and

2 c. Immobilizing CFT discharge isolation valves in their closed positions.

The Reference 3 analyses demonstrate the PORV can maintain RCS pressure below limits when makeup flow capability is restricted. Consequently, the administrative controls require makeup flow capability be limited in the LTOP MODES.

Since the PORV cannot do this for one HPI pump, or discharging the CFTs, the LCO also requires the HPI
2 actuation circuits deactivated and the CFTs isolated.

(continued)

BASES

2 APPLICABLE SAFETY ANALYSES (continued)

The isolated CFTs must have their discharge valves closed and the valve power breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at 325°F. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years (EFPYs) of operation for Units 1 and 3, and 19 EFPY for Unit 2.

This LCO will deactivate the HPI actuation when the RCS temperature is $\leq$ 325°F.

Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORV is set to open at $\leq$ 480 psig. The setpoint is derived by modeling the performance of the LTOP system for different LTOP events. The PORV setpoint at or below the derived limit ensures the Reference 1 limits will be met.

The PORV setpoint is re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement induced by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations.

The PORV is considered an active component. Therefore, its failure represents the worst case LTOP single active failure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Administrative Controls Performance

Limiting RCS pressure when RCS temperature is < 220°F provides a minimum margin to the RCS P/T limit. Restricting RCS makeup flow capability and pressurizer level and controls on the use of high pressure nitrogen limit the pressurization rate during an LTOP event. Alarms ensure early operator recognition of the occurrence of an LTOP event. The combination of minimum margin to the limit, limited pressurization rate and OPERABLE alarms ensure ten minutes are available for operator action to mitigate an LTOP event.

The LTOP System satisfies Criterion 2 and Criterion 3 of 10 CFR 50.36 (Ref.6).

LCO

The LCO requires an LTOP System OPERABLE with a limited coolant input capability and a pressure relief capability. The LCO requires HPI to be deactivated and the CFTs to be isolated. For pressure relief, it requires the pressurizer coolant at or below a maximum level and the PORV OPERABLE with a lift setting at the LTOP limit, with other specified administrative controls.

2
2

The pressurizer is OPERABLE with a coolant level limited such that $\geq$ 10 minutes are available for operator action to mitigate the consequences of an LTOP event.

The PORV is OPERABLE when its block valve is open, its lift setpoint is set at $\leq$ 480 psig and testing has proven its ability to open at that setpoint, and power is available to the two valves and their control circuits.

APPLICABILITY

This LCO is applicable in MODE 3 when any RCS cold leg temperature is $\leq$ 325°F, and in MODES 4, 5 and 6 when an RCS vent capable of mitigating the most limiting LTOP event is not open. The Applicability temperature of 325°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above 325°F. With the vessel head off, overpressurization is not possible. With an RCS vent capable of mitigating the most limiting LTOP event open, an

(continued)

BASES

APPLICABILITY (continued)

LTOP event (including HPI actuation or CFT discharge) is incapable of pressuring the RCS above the RCS P/T limits.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3 above 325°F.

The Applicability is modified by two Notes. Note 1 states
2 that CFT isolation is only required when the CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This Note permits the CFT discharge valve surveillance performed only under these pressure and temperature conditions.

Note 2 permits the PORV to be inoperable when no HPI pumps are running and RCS pressure is < 100 psig. PORV operability is not required when RCS pressure is < 100 psig and HPI pumps are not operating since credible LTOP events progress relatively slowly, thus giving the operator ample time to respond.

ACTIONS

A.1

With the HPI activated, immediate actions are required to deactivate HPI. Emphasis is on immediate deactivation because inadvertent injection with one or more HPI pump OPERABLE is the event of greatest significance, since these events cause the greatest pressure increase in the shortest time.

The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.

B.1, C.1, and C.2

2 An unisolated CFT requires isolation within 1 hour only when the CFT pressure is at or more than the maximum RCS pressure for the existing temperature allowed in LCO 3.4.3.

2 If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide

(continued)

BASES

ACTIONS

B.1, C.1, and C.2 (continued)

two options, either of which must be performed in 12 hours. By placing the unit in MODE 4 with the RCS temperature > 233°F, the CFT pressure of 650 psig cannot exceed the LTOP limits if both tanks are fully injected. Depressurizing the CFTs below the LTOP limit of 373 psig also prevents exceeding the LTOP limits in the same event.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering judgement indicating that a limiting LTOP event is not likely in the allowed times.

D.1, E.1, and E.2

With the PORV inoperable, overpressure relieving capability is lost, and restoration of the PORV within 1 hour is required.

If restoration cannot be completed within 1 hour, either Required Action E.1 or Required Action E.2 must be performed. Required Action E.1 requires increasing RCS temperature within 23 hours to exit the Applicability of the specification. With RCS temperature > 325°F, the CFTs are not required to be isolated. Required Action E.2 requires the RCS be depressurized to less than 100 psig within 35 hours. With reactor pressure < 100 psig more time is available for operator action to mitigate an LTOP event.

2

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in these times.

F.1 and G.1

With Administrative Controls that assure ≥ 10 minutes are available to mitigate the consequences of an event not implemented, the capability for operator action to mitigate an LTOP event may be lost. In this circumstance, compensatory measures must be established to monitor for initiation of an LTOP event. Establishing a dedicated operator within 4 hours to monitor for initiation of an LTOP

(continued)

BASES

ACTIONS

F.1 and G.1 (continued)

event is sufficient to compensate for inoperability of makeup flow restrictions, inoperability of required alarms, or deviation from pressure, temperature or level limits. Establishing a dedicated operator is not sufficient to compensate for not deactivating HPI or isolating CFTs. If the Required Action and associated Completion Time of Condition F is not met, the RCS must be depressurized and an RCS vent path capable of mitigating the most limiting LTOP event must be established within 12 hours. These Completion Times also consider that these activities can be accomplished in these time periods. A limiting LTOP event is not likely in these periods.

2

H.1

With administrative controls which assure ≥ 10 minutes are available to mitigate the consequences of an LTOP event not implemented and the PORV inoperable; or the LTOP System inoperable for any reason other than cited in Condition A through G, Required Action H.1 requires the RCS depressurized and vented within 12 hours.

One or more vents may be used. A vent path capable of mitigating the most limiting LTOP event is specified. Because makeup may be required, the vent size accommodates inadvertent full makeup system operation. Such a vent keeps the pressure from full flow of the makeup pump(s) with a wide open makeup control valve within the LCO limit.

The Completion Time is based on operating experience that these activity can be accomplished in this time period and on engineering judgement indicating that a limiting LTOP transient is not likely in this time.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

Verifications must be performed that HPI is deactivated, and the CFTs are isolated. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP System. The Surveillances are required at 12 hour intervals.

2

(continued)

(A1) (except as marked) Specification 3.4.12

Supp. 2

Action A
Action B
Action D
Action E

5. a. If one or more HPI trains or CFTs are not deactivated, the HPI trains and CFTs shall be deactivated immediately

(A27) isolated (A28) action initiated & deactivate (L16) within one hour (M53) within one hour

b. If the PORV is inoperable, the PORV shall be returned to operable status or the RCS shall be heated above 325°F within (24) hours, or within (36) hours the RCS shall be depressurized to < 100 psig and HPI shall be removed from service.

(23) (M53) (35) (M53) (A15)

Action F
Action G

c. If the second LTOP train is inoperable, the second train shall be restored to operable status or compensatory measures shall be provided to monitor for initiation of an LTOP event within 4 hours, or within 16 hours the RCS shall be depressurized and a vent path capable of mitigating the most limiting LTOP event shall be opened.

Bases - Units 1, 2 and 3

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, startup and shutdown operations, and inservice leak and hydrostatic tests. The various categories of load cycles used for design purposes are provided in Table 5.2-1 of the FSAR.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1699 and BAW-1697.

(A2)

The Figures specified in 3.1.2-1, 3.1.2-2 and 3.1.2-3 present the pressure-temperature limit curves for normal heatup, normal cooldown and hydrostatic tests respectively. The limit curves are applicable up to the indicated effective full power years of operation. These curves will be adjusted to include possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations.

The cooldown limit curves are not applicable to conditions of off-normal operation (e.g., small LOCA and extended loss of feedwater) where cooling is achieved for extended periods of time by circulating water from the HPI through the core. If core cooling is restricted to meet the cooldown limits under other than normal operation, core integrity could be jeopardized.

The pressure-temperature limit lines shown on the figures specified in 3.1.2-1 for reactor criticality and on the figures referred to in 3.1.2-3 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice hydrostatic testing.

Add Action C (L16)

Add Action H (M29)

Amendment No. 204 (Unit 1)
Amendment No. 204 (Unit 2)
Amendment No. 201 (Unit 3)

A25 A CTS provision comparable to the Note to ITS 3.4.15 Actions does not exist. The Note to ITS 3.4.15 Actions states that ITS 3.0.4 is not applicable. This Note therefore permits entry into MODES 1, 2, 3 and 4 when not meeting LCO 3.4.15. CTS does not preclude entry into an operational condition when all the requirements for being in the operational condition are not met. Therefore this change is administrative in nature and is consistent with the NUREG.

A26 CTS 3.1.3.1 requires the reactor coolant temperature to be > 525°F. ITS LCO 3.4.2 requires the reactor coolant temperature to be ≥ 525°F. This difference is so slight as to be imperceptible. Reactor criticality at precisely 525°F is within the applicable analysis. Therefore this, change is considered to be administrative and is consistent with the NUREG.

A27 CTS 3.1.2.9.2.c requires CFTs be deactivated. The Bases for CTS 3.1.2 (page 3.1-4b) states that the CFTs may be deactivated by shutting valves CF-1 and CF-2, tagging open the valve breaker and tagging the valves in the closed position or by reducing CFT pressure to less than the maximum permissible RCS pressure for existing RCS temperature. ITS 3.4.12 LCO Applicability Note 1 specifies that CFT isolation is not required when CFT pressure is less than the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in ITS Specification 3.4.3. ITS LCO 3.4.12, Action B and SR 3.4.12.2 refer to the CFT as "isolated" rather than "deactivated." Since ITS 3.4.12 LCO Applicability Note 1 removes the need for defining "deactivated" and the term "isolated" is equivalent to "deactivated" as it relates to closing valves, this change is a presentation preference which is administrative in nature. The change is consistent with the NUREG.

A28 CTS 3.1.2.9.5.a requires HPI pumps be deactivated immediately. ITS 3.4.12 Actions A require action be initiated immediately to deactivate HPI pumps. The intent of the ITS is the same, however, the ITS wording recognizes the literal impossibility of completing physical actions as may be required by a strict interpretation of "immediately" while still requiring timely completion. An ITS specified Required Action with an immediate completion time is required by ITS 1.3, Completion Times, to be pursued without delay and in a controlled manner. Therefore, this change is considered to be an administrative change and is consistent with the NUREG.

A29 CTS figures 3.1.2-3A, 3.1.2-3B and 3.1.2-3C are titled, "Reactor Coolant System Inservice Leak and Hydrostatic Test Heatup and Cooldown Limitations . . . ." The equivalent ITS figures, 3.4.3-3, 3.4.3-6 and 3.4.3-9 are titled, "Reactor Coolant System Leak and Hydrostatic Test Heatup and Cooldown Limitations . . . ." The term "inservice" is not necessary since the tables are also valid for non-inservice leak tests. The elimination of the term "inservice" from the title is considered an administrative change which preserves the intent of the CTS.

M24 A CTS requirement comparable to ITS 3.4.11 RA C.1 does not exist. With gross specific activity not within limits, CTS 3.0 requires the unit be made subcritical. With gross specific activity not within limits, ITS RA C.1 requires the unit be placed in MODE 3 with RCS average temperature < 500°F. This change provide appropriate actions to limit the time period for operation when RCS activity is not within limits. The ITS requirement to place the unit in MODE 3 with RCS average temperature < 500°F is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M25 Not used.

M26 Not used.

M27 Not used.

M28 Not used.

M29 A CTS requirement comparable to ITS 3.4.12 Action H does not exist. With administrative controls not implemented and the PORV inoperable or with the LTOP inoperable for other reasons, CTS requirements are specified in CTS 3.1.2.9.5.a, b and c as well as CTS 3.0. The most restrictive of these requirements is CTS 3.1.2.9.5.c which requires measures be provided to monitor for initiation of an LTOP event within 4 hours or the RCS be depressurized and a vent path capable of mitigating the most limiting LTOP event be established within 16 hours. For this Condition, ITS 3.4.12 Action H requires the RCS be depressurized and a vent path capable of mitigating the most limiting LTOP event be established within 12 hours. This change requires placing the unit in a MODE where the specification is no longer applicable. The requirement to depressurize and establish the specified vent path within 12 hours is a more restrictive requirement on unit operation and is consistent with the NUREG.

M30 CTS requirements comparable to ITS SRs 3.4.12.1, SR 3.4.12.2, SR 3.4.12.3, SR 3.4.12.4, SR 3.4.12.5, and SR 3.4.12.6 do not exist. ITS SR 3.4.12.1 requires periodic verification that HPI is deactivated.
2 ITS SR 3.4.12.2 requires periodic verification that CFTs are isolated. ITS SR 3.4.12.3 requires periodic verification that pressurizer level is within limits. ITS SR 3.4.12.4 requires periodic verification that an PORV block valve is open. SR 3.4.12.5 requires a periodic channel functional test for the PORV. ITS SR 3.4.12.6 requires periodic verification that administrative controls other than limits on pressurizer level are implemented. These changes are necessary to periodically confirm the LCO requirements are met and ensures timely identification when LCO requirements are not met. These changes are additional restrictions upon unit operation and are consistent with the NUREG.

M31 CTS requirements comparable to ITS 3.4.13 Action A and RA B.2 does not exist for CTS 3.1.6.2 which is the CTS requirement comparable to ITS LCO

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with ~~a maximum of [one]~~ makeup pump capable of injecting into the RCS, high pressure injection (HPI) deactivated, and the core flood tanks (CFTs) isolated and:

a. ~~Pressurizer~~ level ≤ [220] inches and ~~an OPERABLE~~ power operated relief valve (PORV) with a lift setpoint of ≤ [555] psig; or

b. ~~The~~ RCS ~~depressurized~~ and an RCS vent of ≥ ~~[0.75]~~ square inch.

APPLICABILITY: ~~MODE 4~~ when any RCS cold leg temperature is ≤ ~~[283]~~ °F,
~~MODE 5~~,
~~MODE 6 when the reactor vessel head is on.~~

NOTES

1. CFT isolation is only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in ~~the PTLR~~.

MODES 4, 5 and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. More than [one] makeup pump capable of injecting into the RCS.	A.1 --------NOTE-------- Two makeup pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes. ---------- Initiate action to verify only [one] makeup pump is capable of injecting into the RCS.	Immediately	
B. HPI activated.	B.1 Initiate action to verify HPI deactivated.	Immediately	3.1.2.9.5.a
C. A CFT not isolated when CFT pressure is greater than or equal to the maximum RCS pressure for existing temperature allowed [in the PTLR] by Specification 3.4.3.	C.1 Isolate affected CFT.	1 hour	3.1.2.9.5.a
D. Required Action C.1 not met within the required Completion Time.	D.1 Increase RCS temperature to > [178]°F.	12 hours	DOC L16
OR C. of Condition B not met	D.2 Depressurize affected CFT to < [555] psig.	12 hours	DOC L16

(continued)

Administrative controls that assure ≥10 minutes are available to mitigate the consequences of an LTOP event Not implemented and PORV Inoperable

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~Pressurizer level > [220] inches.~~ ~~AND~~ ~~PORV inoperable.~~ ~~OR~~ LTOP System inoperable for any reason other than Condition A through Condition ~~H~~ G.	I.1 ~~Restore~~ LTOP System ~~to OPERABLE status.~~ ~~OR~~ I.2 Depressurize RCS and establish RCS vent ~~of [0.75] square inch~~ path capable of mitigating the most limiting LTOP event.	1 hour 12 hours

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Doc M29

27

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.12.1 Verify a maximum of ~~[one]~~ makeup pump is capable of injecting into the RCS.	~~12 hours~~ 27
SR 3.4.12.2 Verify HPI is deactivated.	12 hours
SR 3.4.12.3 Verify each CFT is isolated.	12 hours

Doc M30

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Supp. 2

(continued)

elimination of the reference to emergency power supply eliminates potential confusion which could result since each pressurizer heater can be supplied from emergency power. Retaining the reference to emergency power supplies would imply that some pressurizer heaters are not capable of being supplied from an emergency power source.

23 Not used.

24 When RCS cold leg temperature is ≤ 325°F, Low Temperature Overpressure Protection (LTOP) is applicable (ITS LCO 3.4.12) and provides appropriate requirements for pressure control. The NUREG LCO 3.4.10 Applicability and associated Note are modified to eliminate MODE 4 and eliminate Applicability in MODE 3 when LCO 3.4.12 is applicable. NUREG 3.4.10 RA B.2 is modified to require placing the unit in a condition outside the modified Applicability.

25 Not used.

26 Consistent with the current licensing basis (CLB), NUREG Specification 3.4.11, Pressurizer PORV is not adopted. Generic Letter 90-06 recommended adoption of similar Technical Specification requirements. Duke Power Company (DPC) in its responses to GL 90-06 dated December 20, 1990, and March 27, 1991 concluded the PORV did not satisfy the criteria of the Interim Policy Statement on Technical Specification Improvements. By its letter dated June 9, 1994, the NRC found the DPC response to the recommendation to be acceptable based on plant specific arguments.

27 LCO 3.4.12 is modified to retain consistency with the CLB. LCO 3.4.12 is modified to include CTS allowed flexibility regarding limits on
2 pressurizer level and makeup flow. Current requirements for the PORV and administrative controls are retained.

The portion of NUREG LCO 3.4.12 regarding the RCS vent is not adopted. The Applicability for LCO 3.4.12 is modified to reflect the ONS LTOP enable temperature is 325°F and occurs in ITS MODE 3. Additionally, the Applicability for LCO 3.4.12 is modified to reflect the CLB regarding a vent path capable of mitigating the most limiting event. With a vent path capable of mitigating the most limiting LTOP event, no other LTOP requirements are necessary since the vent path precludes an LTOP event from causing RCS pressure to exceed the P/T limits. NUREG SR 3.4.12.1 is not adopted since the number of makeup pumps is not limited. NUREG SR 3.4.12.6 is not adopted since the RCS vent path is not a part of the associated LCO.

NUREG 3.4.12 Action A, which provide Required Actions when more than the specified number of makeup pumps are capable of injecting into the RCS; and NUREG Actions E and F which provide Required Actions when pressurizer level is greater than the specified limit are not adopted.

BASES

BACKGROUND (continued)

The LTOP approach to protecting the vessel by limiting coolant addition capability ~~allows a maximum of one makeup pump~~, and requires deactivating HPI, and isolating the core flood tanks (CFTs).

Should ~~more than one~~ HPI pump inject on an HPI actuation the pressurizer level and PORV ~~or another RCS vent cannot~~ prevent overpressurizing the RCS. ~~Even with only one HPI pump OPERABLE, the vent cannot prevent RCS overpressurization.~~

The pressurizer level ~~limit~~ provides a compressible vapor space or cushion (either steam or nitrogen) that can accommodate a coolant insurge and prevent a rapid pressure increase, allowing the operator time to stop the increase. The PORV, with reduced lift setting, ~~or the RCS vent~~ is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI deactivated, the ability to provide RCS coolant addition is restricted. To balance the possible need for coolant addition, the LCO does not require the ~~M~~akeup ~~S~~ystem to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the ~~M~~akeup ~~S~~ystem can provide flow with the ~~OPERABLE~~ makeup ~~pump~~ through the makeup control valve.

PORV Requirements

As ~~designed~~ for the LTOP ~~system~~, ~~each~~ PORV is signaled to open if the RCS pressure approaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the PORV is signaled to open. Maintaining the setpoint within the limits of the LCO ensures the Reference 1 limits will be met in any event analyzed for LTOP.

When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases ~~coolant~~, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and adding nitrogen to the pressurizer.

HPI actuation and CFT discharge are the transients that result in exceeding P/T limits within < 10 minutes, in which time no operator action is assumed to take place. In the rest, operator action after that time precludes overpressurization. The analyses demonstrate that the time allowed for operator action is adequate, or the events are self limiting and do not exceed P/T limits.

starting an RCP and adding nitrogen to the pressurizer are self limiting events.

The following are required during the LTOP MODES to ensure that transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. ~~Deactivating all but [one] makeup pump~~;

limiting RCS makeup flow capability

b. Deactivating HPI; and

c. Immobilizing CFT discharge isolation valves ~~in their closed positions~~.

Supp 2

administrative controls require makeup flow capability be limited

The Reference 3 analyses demonstrate the PORV can maintain RCS pressure below limits when ~~only one makeup pump is actuated~~. Consequently, the LCO allows only ~~one~~ makeup ~~pump to be OPERABLE~~ in the LTOP MODES.

makeup flow capability is restricted

Since the PORV cannot do this for one HPI pump ~~and the RCS vent cannot do this for even one pump,~~ the LCO also requires the HPI actuation circuits deactivated and the CFTs isolated.

Supp 2

or discharging the CFTs

The isolated CFTs must have their discharge valves closed and the valve power breakers fixed in their open positions. ~~The analyses show the effect of CFT discharge is over a narrower RCS temperature range (175°F and below) than that of the LCO ([283]°F and below).~~

Fracture mechanics analyses established the temperature of LTOP Applicability at ~~[283]~~ 325°F. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years (EFPYs) of operation

for Units 1 and 3, and 19 EFPY for Unit 2

(continued)

Insert B3.4-61A

Limiting RCS pressure when RCS temperature is < 220°F provides a minimum margin to the RCS P/T limit. Restricting RCS makeup flow capability and pressurizer level and controls on the use of high pressure nitrogen limit the pressurization rate during an LTOP event. Alarms ensure early operator recognition of the occurrence of an LTOP event. The combination of minimum margin to the limit, limited pressurization rate and OPERABLE alarms ensure ten minutes are available for operator action to mitigate an LTOP event.

Insert B3.4-61B

2 The LCO requires HPI to be deactivated and the CFTs to be isolated.

Insert B3.4-61C

limited such that ≥ 10 minutes are available for operator action to mitigate the consequences of an LTOP event.

BASES

LCO (continued)

For the depressurized RCS, an RCS vent is OPERABLE when open with an area of at least [0.75] square inches.

Supp. 1

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ [283] °F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The Applicability temperature of [283] °F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above [283] °F. With the vessel head off, overpressurization is not possible.

and in MODES 4, 5, & 6 when an RCS vent capable of mitigating the most limiting LTOP event is not open.

325

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3 and MODE 4 above [283] °F.

Insert B3.4-62A

The Applicability is modified by two Notes. Note 1 states that CFT isolation is only required when the CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This Note permits the CFT discharge valve surveillance performed only under these pressure and temperature conditions.

Supp. 2

Supp. 1

Insert B3.4-62B

ACTIONS

A.1 and B.1

With two or more makeup pumps capable of injecting into the RCS or if the HPI is activated, immediate actions are required to render the other pump(s) inoperable or to deactivate HPI. Emphasis is on immediate deactivation because inadvertent injection with one or more HPI pump OPERABLE is the event of greatest significance, since these events cause the greatest pressure increase in the shortest time. Also, the vent cannot mitigate overpressurization from the injection of even one HPI pump.

The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.

Required Action A.1 is modified by a Note that permits two pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.

(continued)

BASES

ACTIONS (continued)

Supp 2

C.1, C.1, and C.2

An unisolated CFT requires isolation within 1 hour only when the CFT pressure is at or more than the maximum RCS pressure for the existing temperature allowed in LCO 3.4.3.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in 12 hours. By increasing the RCS temperature > 225°F, the CFT pressure of 600 psig cannot exceed the LTOP limits if both tanks are fully injected. Depressurizing the CFTs below the LTOP limit of 554 psig also prevents exceeding the LTOP limits in the same event.

Placing the unit in Mode 4 with

650

233

373

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that a limiting LTOP event is not likely in the allowed times.

judgement

E.1, F.1, and F.2

With the pressurizer level more than [220] inches, the time for operator action in a pressure increasing event is reduced. The postulated event most affected in the LTOP MODES is failure of the makeup control valve, which fills the pressurizer relatively rapidly. Restoration is required within 1 hour.

If restoration within 1 hour in either case cannot be accomplished, Required Actions F.1 and F.2 must be performed within 12 hours to close the makeup control valve and its isolation valve. These Required Actions limit the makeup capability, which is not required with a high pressurizer level, and permit cooldown and depressurization to continue. Heatup must be stopped because heat addition decreases the reactor coolant density and increases the pressurizer level.

The Completion Times again are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that a limiting LTOP transient is not likely in the allowed times.

(continued)

Insert B3.4-64A

Required Action E.1 requires increasing RCS temperature within 23 hours to exit the Applicability of the specification. With RCS temperature > 325°F, 2 the CFTs are not required to be isolated. Required Action E.2 requires the RCS be depressurized to less than 100 psig within 35 hours. With reactor pressure < 100 psig more time is available for operator action to mitigate an LTOP event.

Insert B3.4-64B

F.1 and G.1

With Administrative Controls that assure ≥ 10 minutes are available to mitigate the consequences of an event not implemented, the capability for operator action to mitigate an LTOP event may be lost. In this circumstance, compensatory measures must be established to monitor for initiation of an LTOP event. Establishing a dedicated operator within 4 hours to monitor for initiation of an LTOP event is sufficient to compensate for inoperability of makeup flow restrictions, inoperability of required alarms, or deviation from pressure, temperature or level limits. Establishing a dedicated operator is 2 not sufficient to compensate for not deactivating HPI or isolating CFTs. If the Required Action and associated Completion Time of Condition F is not met, the RCS must be depressurized and an RCS vent path capable of mitigating the most limiting LTOP event must be established within 12 hours. These Completion also Times consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in these periods.

BASES

ACTIONS

E.1 and E.2 (continued)

The PORV has a larger area and may be used for venting by opening and locking it open.

This size RCS vent or the PORVs a vent cannot maintain RCS pressure below LTOP limits if the HPI and CFT systems are inadvertently actuated. Therefore, verification of the deactivation of two HPI pumps, HPI injection, and the CFTs must accompany the depressurizing and venting. Since these systems are required deactivated by the LCO, SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 require verification of their deactivated status every 12 hours.

Again, The Completion Times are based on operating experience that these activities can be accomplished in this these time periods and on engineering evaluations indicating that a limiting LTOP transient is not likely in these times. Judgement this

SURVEILLANCE REQUIREMENTS

Supp 2

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

Verifications must be performed that only [one] makeup pump is capable of injecting into the RCS, the HPI is deactivated, and the CFT discharge isolation valves are closed and immobilized. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP System. The Surveillances are required at 12 hour intervals.

The 12 hour intervals are shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation within the safety analysis.

SR 3.4.12.4

Verification of the pressurizer level at ~~[220] inches~~ by observing control room or other indications ensures a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients.

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level

is less than the volume necessary to assure ≥ 10 minutes are available for operator action to mitigate an LTOP event.

(continued)

ITS Section 3.4

ID 83

Subject: Revised 3.4.13 to reinstate NUREG terminology of "of Condition A."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 300 gallon per day total primary to secondary LEAKAGE through all steam generators (SGs); and

e. 150 gallon per day primary to secondary LEAKAGE through any one SG.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours 36 hours

2

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE; 3.1.6.3

b. 1 gpm unidentified LEAKAGE; 3.1.6.2

c. 10 gpm identified LEAKAGE; 3.1.6.1

d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and DOC A4 3.1.6.4

e. 4720 gallons per day primary to secondary LEAKAGE through any one SG. (1)

(4) 300 gallons per day → 150

Supp 1

APPLICABILITY: MODES 1, 2, 3, and 4. Table 4.1-2 Footnote (2)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours	3.1.6.4
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	(5) (12) (6) hours 36 hours	3.1.6.1 3.1.6.2 3.1.6.3 3.1.6.4 3.1.6.3 3.1.6.4

Supp 2

ITS Section 3.4

ID 125

Subject: Removed JFD 17 annotation and replace with TSTF 262 on NUREG page 3.4-9.

TSTF-153

Not be in operation

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops – MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and ~~at least~~ one loop shall be in operation.

Doc M13

TSTF-261

Supp. 1

----------NOTE----------

1. All reactor coolant pumps (RCPs) may ~~be de-energized~~ for ≤ 8 hours per 24 hour period for the transition to or from the DHR System, and all RCPs and DHR pumps may be de-energized for ≤ 1 hour per 8 hour period for any other reason, provided:

Doc M13

a. No operations are permitted that would cause reduction of the RCS boron concentration; and 3.1.1.a.2

b. Core outlet temperature is maintained at least 10°F below saturation temperature. Doc M13

Insert 3.4-9A

TSTF 262

Supp 2

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ~~RCS~~ loop inoperable. AND Two DHR loops inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately

Doc M13

TSTF 263

Supp. 1

(continued)

AND

A.2 ---Note--- Only required if DHR loop is OPERABLE.

Be in MODE 5. 24 hours

ITS Section 3.4

ID 127

Subject: Revised Section 3.4 DOC M1 to refer to the "Applicability Note" instead of the "LCO Note."

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS requirements comparable to ITS 3.4.1 do no exist. ITS LCO 3.4.1 requires RCS DNBR parameters for loop pressure, loop average temperature and RCS total flow to be within the limits specified in the COLR. The
2 Note to the Applicability states limits on loop pressure do not apply during specified changes in THERMAL POWER. ITS Specification 3.4.1 specifies an Applicability of MODE 1. The ITS LCO 3.4.1 Actions require restoring DNBR parameters to within limits within 2 hours or exiting the Applicability for the Specification within 12 additional hours. ITS SR 3.4.1.1, SR 3.4.1.2 and SR 3.4.1.3 require verification that each DNBR parameter is within limit at a 12 hour frequency. ITS SR 3.4.1.4 requires verification by measurement that total RCS flow is within limit at an 18 month Frequency. Specification 3.4.1 ensures limits on RCS pressure, temperature, and flow rate are met "to ensure that the core operates within the limits assumed for the plant safety analyses." Operating within these limits will result in meeting departure from nucleate boiling ratio (DNBR) criteria in the event of a DNB limited transient. The addition of ITS 3.4.1 requirements is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M2 CTS 3.1.3.1 requires the limit for the minimum temperature for criticality to be applied to the reactor coolant but does not explicitly require the limit be applied to each RCS loop individually. ITS 3.4.2 explicitly requires the limit be applied individually to each RCS loop, since it is possible for an individual RCS loop average temperature to be out of limit even though the overall RCS average temperature remains within the limit. Requiring the limit be applied to each loop's average temperature is necessary to ensure both loops are within limits. This change is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M3 A CTS requirement comparable to ITS SR 3.4.2.1 does not exist. ITS SR 3.4.2.1 requires periodic verification that the reactor coolant loop average temperatures are within specified limits if any RCS loop is $< 530°F$. This change is necessary to periodically confirm the LCO requirements are met and ensures timely identification when LCO requirements are not met. The addition of ITS SR 3.4.2.1 is an additional restriction upon unit operation and is consistent with the NUREG.

M4 A CTS requirement comparable to ITS SR 3.4.3.1 does not exist. ITS SR 3.4.3.1 requires periodic verification that the RCS pressure and temperature are within limits during heatup or cooldown operations or leak or hydrostatic testing. This change is necessary to periodically confirm the LCO requirements are met during the specified evolutions and ensures timely identification when LCO requirements are not met in other than MODES 1, 2, 3 or 4. The addition of ITS SR 3.4.3.1 is an additional restriction upon unit operation and is consistent with the NUREG.

ITS Section 3.4

ID 138

Subject: Revised Applicability statements for LCO 3.4.14 to provide correct indentation.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limits.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 except valves in the decay heat removal (DHR) flow
2 path when in, or during the transition to or from, the
2 DHR mode of operation.

ACTIONS

----------NOTES----------
1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	----------NOTE---------- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 and be on the RCS pressure boundary or the high pressure portion of the system. ----------	(continued)

ITS Section 3.4

ID 109

Subject: Revised SR 3.4.13.1 to include changes resulting from TSTF 116, R2 and to remove method of performing SR (RCS water inventory balance).

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	----------NOTE---------- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify RCS Operational Leakage is within limits.	72 hours
SR 3.4.13.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

2

BASES

ACTIONS

B.1 and B.2 (continued)

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits ensures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and may be positively identified by inspection. Unidentified LEAKAGE is determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is measured
2 by effluent monitoring within the secondary systems or
2 comparison of primary and secondary radioisotope
2 concentrations. These methods provide the required leakage
2 detection sensitivity to ensure leakage is within limits..

The RCS water inventory balance must be performed with the
2 reactor at steady state operating conditions (stable
2 temperature, power level, pressurizer and makeup tank
2 levels, makeup and letdown [and RCP seal injection and
2 return flows]). Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. This 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP pump seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. UFSAR, Section 3.1.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.

Specification 3.4.13

(A1) (except as marked)

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rod Movement [1]	Movement of Each Rod	Monthly (SEE 3.1)
2. Pressurizer Safety Valves	Setpoint	18 months[4] (SEE 3.4.10)
3. Main Steam Safety Valves	Setpoint	18 months[4] (SEE 3.7)
4. Refueling System Interlocks[5]	Functional	Prior to Refueling (SEE 3.3)
5. Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly (SEE 3.7)
6. Reactor Coolant System [2] Leakage	Evaluate (M35) Verify RCS operational leakage is within limits	Daily (72 hrs) (L14)
7. Emergency Condenser [6] Circulating Water System Test	Functional	18 months
8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly (SEE 3.7)
9. Spent Fuel Cooling System	Functional	Prior to Refueling
10. High Pressure and Low [3] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing (SEE 3.5)
11. Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months
12. MSLB Feedwater Isolation [7] Feature	Functional	18 months (SEE 3.7)
13. Essential Siphon Vacuum [8] System Test	Functional	Quarterly

Supp 2 — SR 3.4.13.1

Supp 2

Supp 2

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

TSC 95-03
3 of 4

If LEAKAGE through the tubes of one steam generator exceeds specified limits, CTS 3.1.6.4 requires a reactor shutdown be initiated within 4 hours and the reactor be in cold shutdown within 36 hours. Initiation of reactor shutdown within 4 hours implies 4 hours to restore LEAKAGE to within limits. With RCS LEAKAGE not within limits, ITS 3.4.13 Action A permits 4 hours to restore LEAKAGE to within limits. With the Required Action and associated Completion Time of Condition A not met, ITS 3.4.13 Action B requires the unit be in MODE 3 within 12 hours and MODE 5 within 36 hours. The requirement to place the unit in MODE 3 within 12 hours is provided to ensure unit cooldown to MODE 5 is begun promptly to place the unit in a condition where the driving force (RCS pressure) for RCS leakage is minimized. The requirement to place the unit in MODE 3 within 16 hours after entry into ITS 3.4.13 Action A is a more restrictive requirement upon unit operation. These changes are consistent with the NUREG.

M34 CTS 3.1.6.5 permitted the rate of unit shutdown and the condition of shutdown to be based on a case specific safety evaluation for reactor shutdowns required by CTS 3.6.1.1, 3.6.1.2 and 3.6.1.3. ITS 3.4.13 Actions A and B provide generic requirements for Conditions involving excessive RCS leakage comparable to CTS 3.6.1.1, 3.6.1.2 and 3.6.1.3.

The requirement to place the unit in MODE 3 within 12 hours and in MODE 5 within 36 hours provides for promptly placing the unit in a condition where the driving force (RCS pressure) for RCS leakage is minimized. The completion times are reasonable, based on operating experience to achieve the require conditions in an orderly manner, without challenging unit systems. The elimination of the flexibility afforded by case specific safety evaluations and its replacement with prescription Conditions and Completion Times is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M35 CTS Table 4.1-2 Item 6 requires RCS LEAKAGE be evaluated daily. ITS
2 SR 3.4.13.2 requires verification that RCS operational LEAKAGE limits
2 are within limits. The explicit requirement to verify RCS leakage is
2 within limits, is consistent with current practice and does not result
2 in an unacceptable burden upon unit operation. The prescriptive nature of SR 3.4.13.1 is considered to be a more restrictive requirement upon unit operation and is consistent with the NUREG.

M36 CTS 3.1.6.10.b specifies a 150 psig minimum pressure differential for PIV testing. ITS SR 3.4.14.1 leakage limit is specified leakage at an RCS pressure of $\geq$ 2150 psia and $\leq$ 2190 psia. This results in requiring leakage limits to be met at a substantially greater differential pressure (the difference between 2150 psig and the pressure rating for the associated low pressure system). Although the PIV leakage may be measured at a lower pressure differential, the measured leakage must be adjusted to determine the leakage at the specified RCS pressure. The requirement to adjust measured PIV leakage for actual test conditions is necessary to provide results applicable to normal RCS pressure conditions. The requirement to determine leakage at the specified RCS

CTS

TSTF 116, R1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.13.1	----------NOTE---------- Not required to be performed ~~in MODE 3 or 4~~ until 12 hours of steady state operation. ---------- after establishment	----NOTE---- ~~Only required to be performed during steady state operation~~ ----------	Table 4.1-2 footnote (2)
	~~Perform~~ RCS ~~water~~ inventory ~~balance~~. SS	72 hours	Table 4.1-2 Item 6
SR 3.4.13.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program	4.17.4.a

Supp 2

Verify RCS operational Leakage is within limits ~~by performance of~~

TSTF 61

SS

concentration, and restrictions on further reductions in RCS volume also apply. Additionally, there is no comparable restriction in CTS. Permitting DHR pumps to not be in operation is acceptable based on the short duration and the other specified limitations.

50 NUREG specification 3.4.16 is renumbered to be ITS 3.4.11. This renumbering permits filling the gap left by the non-adoption of NUREG 3.4.11 (see JFD 26) and reduces renumbering the subsequent specifications.

51 NUREG LCO 3.4.12 is modified to reflect the CLB regarding LTOP requirements. The limit upon the number of makeup pumps capable of providing makeup in not adopted. The ONS LTOP design places restrictions upon RCS makeup flow by installation of a device which limits the opening of the makeup valve. Limiting the opening of the makeup valve limits RCS makeup flow rate.

52 NUREG LCO 3.4.15 part a, Condition A and SR 3.4.15.3 are modified to state sump level indication instead of sump level monitor. The change reflects plant specific terminology since the term sump level monitor is not routinely used. The containment normal sump level indications "monitor" the water level in the containment normal sump but are not referred to as a monitor."

A conforming change is made to NUREG LCO 3.4.15 Condition D. The entry Condition is modified to refer to instrumentation functions instead monitors. Although the radiation monitoring instrumentation monitor is considered a monitor, the sump level indication is not. This change is necessary to reflect plant specific terminology.

53 NUREG LCO 3.4.15 A is modified to state containment normal sump instead of containment sump. The ONS design includes two sumps in the containment, each with sump level indication. The containment normal sump is smaller and provides greater leak detection capability. This change is necessary to discriminate between the containment normal sump level indication and the containment emergency sump level indication.

54 The NUREG 3.4.4 bracketed value of 79.9% RTP for three RCP operation is removed and the plant specific value of 75% RTP is inserted. The value of 75% RTP is the power level assumed in the accident and transient analsyis for three RCP operation.

2 55 NUREG SR 3.4.13.1, as modified by approved TSTF 116, Revision 1,
2 requires that RCS operational leakage be verified within limits by
2 performance of an RCS water inventory balance. The ITS does not specify
2 performance of an RCS water inventory balance since this method is only
2 used for determining unidentified leakage (pressure boundary leakage.
2 At ONS, to provide the required leakage detection sensitivity, primary
2 to secondary leakage is determined by effluent monitoring within the
2 secondary system or comparison of primary and secondary radioisotope

2 concentrations. This method of performing the surveillance is described
2 in the Bases.

INSERT B 3.4-72A (13)

TSTF 116, R2

INSERT B 3.4-72.B

Supp 2

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits ensures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and ~~can only~~ be positively identified by inspection. (1) Unidentified LEAKAGE ~~and identified LEAKAGE~~ are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary ~~steam and feedwater~~ systems. (2)

TSTF 116, R2

(2) Tracy

A Note is added allowing that (13)

The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. Therefore, this SR is not required to be performed ~~in MODES 3 and 4~~ until 12 hours (6) after establishing steady state operation ~~near operating pressures have been established~~.

The 12 hour allowance provides sufficient time to collect & process all necessary data after stable plant conditions are established

Supp 2

Steady state operation (since) is required to perform a proper water inventory balance; calculations during maneuvering are not useful ~~and a Note requires the Surveillance to be met when steady state is established~~. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP pump seal injection and return flows.

TSTF 116, R2

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. ~~The Note states that the SR is required to be performed in steady state operation.~~ (A)

TSTF 116, R2

Supp. 2

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam

(continued)

2 **INSERT B3.4-72A**
2
2 or comparison of primary and secondary radioisotope concentrations. These
2 methods provide the required leakage detection sensitivity to ensure leakage
2 is within limits.
2
2
2
2 **INSERT B3.4-72B**
2
2 (stable temperature, power level, pressurizer and makeup tank levels, makeup
2 and letdown [and RCP seal injection and return flows])

RCS leakage specification. This clarification is based on the definition of RCS pressure boundary specified in 10 CFR 50.2.

10 The NUREG SR 3.4.16.1 description of $\bar{E}$ is modified to agree with the CLB and changes made to the NUREG definition of $\bar{E}$ in ITS Section 1.0.

11 The Bases for NUREG SR 3.4.14.1 is modified to capture information from Footnote (a) to CTS 4.5.1.2.2. This footnote provides details regarding testing methods associated with testing the PIVs. This change is made to retain CTS provisions regarding a permissible testing method for PIV leakage.

12 The Bases is modified to incorporate information relocated from the CTS.

2 13 NUREG SR 3.4.13.1 Bases is modified to specify performance of an RCS
2 water inventory balance as the method for determining unidentified
2 leakage (pressure boundary leakage) and effluent monitoring within the
2 secondary system or comparison of primary and secondary radioisotope
2 concentrations as the method for determining primary to secondary
2 leakage. The latter methods are used to provide the required leakage
2 detection sensitivity.

ITS Section 3.4

ID 123

Subject: Revised NUREG page 3.4-18 Applicability Note to correctly cross reference DOC A31 in place of DOC M21.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ [2475] psig and ≤ [2525] psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > [283] °F.

NOTE
The lift settings are not required to be within the LCO limits for entry into MODE 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for [36] hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1 AND B.2	Be in MODE 3. Be in MODE 4 with any RCS cold leg temperature ≤ [283] °F.	6 hours 12 hours

ITS Section 3.4

ID 124

Subject: Revised ITS 3.4.15 to split out Condition C into Condition C and D due to TSTF 257 being withdrawn.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required containment atmosphere radioactivity monitor inoperable.	B.1.1	Analyze grab samples of the containment atmosphere.	Once per 24 hours
	OR		
	B.1.2	--------NOTE-------- Not required until 12 hours after establishment of steady state operation. ----------	
		Perform SR 3.4.13.1.	Once per 24 hours
	AND		
	B.2	Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days
2 C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 5.	36 hours
2 2 2 2 D. Both required instrument functions inoperable.	D.1	Enter LCO 3.0.3.	Immediately

BASES

ACTIONS

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

be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

The 24 hour interval for SR 3.4.13.1 provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of leak detection is available.

C.1 and C.2

2 If a Required Action of Condition A or B cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

2 D.1

2

2 If both required leakage detection instruments are
2 inoperable, no automatic means of monitoring leakage are
2 available, and immediate plant shutdown in accordance with
2 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 each channel is operating properly. The Frequency of 12 hours is based on

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.2 (continued)

instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of 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 relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown it 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 required 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. Industry operating experience has proven this Frequency is acceptable.

REFERENCES

1. UFSAR, Section 3.1.

2. 10 CFR 50.36.

Specification 3.4.15

Supp 2

A1 (except as marked)

Add Action A — M44

Add Action C — M43

Add RAB 1.1, RAB 1.2 — M41

Add Actions Note — A25

Add Action D — A34

M42

3.1.6 Leakage

Add SR 3.4.15.1, SR 3.4.15.2, SR 3.4.15.3, SR 3.4.15.4

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

(See 3.4.13)

3.1.6.3 If any reactor coolant leakage exists through a non-isolable fault in a RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 If the total leakage through the tubes of any one steam generator equals or exceeds 0.35 gpm, a reactor shutdown shall be initiated within 4 hours and the reactor shall be in a cold condition within the next 36 hours.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2 or 3.1.6.3, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

M43

MODES 1, 2, 3 and 4

Applicability LCO

RAB 2

3.1.6.8 When the reactor is ~~critical and above 2% power~~, two reactor coolant leak detection systems ~~of different operating principles~~ shall be operable, ~~with one of the two systems~~ sensitive to radioactivity. The systems sensitive to radioactivity may be out of service for 48 hours provided two other means to detect leakage are operable.

M44

being contained sump tunnel pairs

30 days

M44

L2

3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7 except that such losses when added to leakage shall not exceed 30 gpm.

(See 3.4.13)

3.1.6.10

a. The maximum allowable leakage for valves CF-12, CF-14, LP-47 and LP-48 shall be as follows:

(See 3.4.14)

1 of 1

A30 CTS 3.1.1.c.1 requires all pressurizer code safety valves to be OPERABLE. ITS 3.4.10 requires two pressurizer safety valves to be OPERABLE. Since the ONS design provides two pressurizer safety relief valves, this change is an administrative change and is consistent with the NUREG.

A31 CTS 3.1.1.c.1 requires pressurizer safety valves to be OPERABLE when the reactor is critical. The Note to ITS LCO 3.4.10 Applicability permits entry into MODE 3 for up to 36 hours with pressurizer lift setting not met for the purpose of setting the lift settings under ambient (hot) conditions. Since CTS does not impose OPERABILITY requirements for the pressurizer safety valves in this operating condition, this change is administrative and is consistent with the NUREG.

A32 CTS 3.1.2.9 provides requirements for LTOP. CTS 3.1.2.9.1.a and 3.1.2.9.1.b specify an Applicability of when one or more RCS cold legs are $\leq$ 325°F and an RCS vent path capable of mitigating the most limiting LTOP event is not open. ITS 3.4.12 specifies an Applicability of MODE 3 when any RCS cold leg temperature is $\leq$ 325°F and also during MODES 4, 5, and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open. The different presentation in the ITS recognizes that an RCS vent path cannot be open in MODE 3 when any RCS cold leg temperature is $\leq$ 325°F or in portions of MODE 4 when RCS temperature is $\geq$ 212°F. This change is administrative and is consistent with the NUREG.

A33 CTS 4.17.4 requires performance of specified SG tube inservice inspections within specified frequencies. ITS SR 3.4.13.2 requires verification of steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program at a Frequency of "In accordance with the Steam Generator Tube Surveillance Program. The details regarding the specified SG tube inservice inspections and specified frequencies are relocated unchanged to ITS 5.5.10, "Steam Generator (SG) Tube Surveillance Program." Therefore this is an administrative change and is consistent with the NUREG.

2 A34 CTS 3.1.6.8 requires two RCS leak detection systems to be OPERABLE when
2 the reactor is critical and above 2% power. No action is provided when
2 both are inoperable. ITS 3.4.15 Required Action D.1 is added to
2 specifically require the unit enter LCO 3.0.3 immediately when both
2 required instrument functions are inoperable. This is equivalent to the
2 CTS since CTS 3.0 requires a shutdown in the event a LCO and/or
2 associated Action requirements cannot be satisfied because of
2 circumstance in excess of those addressed in the specification. There
2 Therefore this is an administrative change and is consistent with the
2 NUREG.

is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M41 CTS requirements comparable to ITS 3.4.15 RA B.1.1 and RA B.1.2 do not exist. With the required containment atmosphere radioactivity monitor inoperable, the actions are specified by either RA B.1.1 or RA B.1.2. ITS 3.4.15 RA B.1.1 requires analyzing grab samples of the containment atmosphere once per 24 hours. ITS 3.4.15 RA B.1.2 requires performing SR 3.4.13.1 (water inventory balance) once per 24 hours. This change provides appropriate measures to partially compensate for the inoperable RCS leakage detection instrumentation. The requirement to either analyze grab samples or perform SR 3.4.13.1 is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M42 CTS requirements comparable to ITS SR 3.4.15.1, SR 3.4.15.2, SR 3.4.15.3 and SR 3.4.15.4 do not exist. ITS SR 3.4.15.1 requires a periodic CHANNEL CHECK of the required containment atmosphere radioactivity monitor. ITS SR 3.4.15.2 requires a periodic CHANNEL FUNCTIONAL TEST of the required containment atmosphere radioactivity monitor. SR 3.4.15.3 requires a periodic CHANNEL CALIBRATION of the required containment sump level indication. ITS SR 3.4.15.4 requires a periodic CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor. This change is necessary to periodically confirm the LCO requirements are met and ensures timely identification when LCO requirements are not met. These additional SRs are more restrictive requirements upon unit operation and are consistent with the NUREG.

M43 CTS 3.1.6.8 requires the RCS leak detection instrumentation to be OPERABLE when the reactor is critical and above 2% power. The Applicability of ITS 3.4.15 is in MODES 1, 2, 3 and 4. A specific CTS provision comparable to ITS 3.4.15 Action C does not exist. With
2 Required Action and associated Completion Time not met, Action C requires placing the unit in MODE 3 within 12 hours and MODE 5 within 36 hours. The requirement for RCS leak detection instrumentation to be OPERABLE in MODES 1, 2, 3, and 4 is appropriate because of elevated RCS pressure and temperature when the unit is in these MODES. The additional requirement for the RCS leak detection instrumentation to be OPERABLE in MODE 1 when reactor power is $\leq$ 2% and MODES 2, 3 and 4 and the associated requirement to place the unit in MODE 3 and MODE 5 are more restrictive requirements upon unit operation and are consistent with the NUREG.

M44 CTS 3.1.6.8 requires two RCS leak detection systems of different operating principles to be operable with one of the two systems being sensitive to radiation. Either the RCS water inventory calculation, or the containment sump level indication could be used as the other leak detection system. ITS LCO 3.4.15 requires the containment sump level indication and a containment atmospheric radiation monitor to be OPERABLE. A CTS requirement comparable to ITS 3.4.15 Action A does not exist. ITS 3.4.15 Action A permits the containment sump level indication to be inoperable for up to 30 days while requiring SR

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.1.2 Perform SR 3.4.13.1.	Once per 24 hours
	AND	
	B.2 Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours
	AND	
	C.2 Be in MODE 5.	36 hours
D. Both required monitors inoperable.	D.1 Enter LCO 3.0.3.	Immediately

Note
Not required until 12 hours after establishment of steady state operation.

TSTF 116, R1

Supp 2

12 → 6 hours → 5

Doc M41

3.1.6.8

Doc M43

Doc M43

instrument functions

52

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required containment atmosphere radioactivity monitor.	92 days

Doc M42

Doc M42

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

acceptable considering the frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

The ~~Required~~ Action ~~A.1 and Required Action A.2~~ are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the ~~monitor is~~ inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE, and required radiation monitors are

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

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

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 leak detection is available.

Required Actions B.1.1, B.1.2, and B.2 are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the containment atmosphere radioactivity monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

C.1 and C.2

If a Required Action of Condition A or B cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

experience, to reach the required ~~plant~~ UNIT conditions from full power conditions in an orderly manner and without challenging ~~plant~~ UNIT systems. (2) (2)

Supp 2

D.1

With both required monitors leakage detection instruments (4) 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 each channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of 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 relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown it 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 required 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. (1)

(continued)

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Specification 3.5.1

LA4

Suppl

4.5.1.1.3 Core Flooding System

a. Every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During pressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

Supp. 1

4.5.1.2 Component Tests

See 3.5.2 + 3.5.3

4.5.1.2.1 Valves - Power Operated

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

Suppl

b. Every 18 months, the following LPI system valves shall be cycled manually to verify the manual operability of these power operated valves:

(1) LPI pump discharge (ES) LP-17,-18
(2) LPI discharge throttling LP-12,-14
(3) LPI discharge header crossover LP-9,-10
(4) LPI discharge to HPI/RBS LP-15,-16

See 3.4.7

4.5.1.2.2 Check Valves

Periodic individual leakage testing[a] of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakage rates and limiting conditions for operation, see Technical Specification 3.1.6.10.

Bases

The Emergency Core Cooling Systems are the principle reactor safety features in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. The HPI system test required by Specification 4.5.1.1.1 verifies that the HPI system responds as required to actuation of ES channels 1 and 2.

A2

See 3.4.7

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

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Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Specification 3.5.2

(A1) <except as marked>

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency	
1.	Control Rod Movement [1]	Movement of Each Rod	Monthly	<SEE 3.1>
2.	Pressurizer Safety Valves	Setpoint	18 months[4]	<SEE 3.4>
3.	Main Steam Safety Valves	Setpoint	18 months[4]	<SEE 3.7>
4.	Refueling System Interlocks[5]	Functional	Prior to Refueling	<SEE 3.3>
5.	Main Steam Stop Valves [1]	Movement of Each Stop ~~Valve~~	Monthly	<SEE 3.7>
6.	Reactor Coolant System [2] Leakage	Evaluate	Daily	<SEE 3.4>
7.	Emergency Condenser [6] Circulating Water System Test	Functional	18 months	<SEE 3.7>
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	
9.	Spent Fuel Cooling System	Functional	Prior to Refueling	
10.	High Pressure and Low [3] Pressure Injection System	Vent Pump Casings <SEE 3.5.3>	31 days ~~Monthly~~ and Prior to Testing	SR 3.5.2.2 (L10)
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	<SEE 3.7>
12.	MSLB Feedwater Isolation [7] Feature	Functional	18 months	
13.	Essential Siphon Vacuum [8] System Test	Functional	Quarterly	

Supp 2

Supp 2

Supp 2

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.5.2

(A1) (except as marked)

(1) Applicable only when the reactor is critical. – (SEE 3.1 + 3.7)

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. (SEE 3.4)

SR 3.5.2.2 Note
(3) Operating pumps excluded.

Supp 2
(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. (SEE 3.4 + 3.7)

(5) Applicable only to the interlocks associated with the Reactor Building Purge System. (SEE 3.3)

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature.

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit. (SEE 3.7)

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Add SR 3.5.2.1 M6

Add SR 3.5.2.6 M7

Specification 3.5.2

Add SR 3.5.2.3 A8

4.5 EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Core Cooling Systems

Applicability

Applies to periodic testing requirements for the Emergency Core Cooling Systems. A1

Objective

To verify that the Emergency Core Cooling Systems are operable.

Specification

4.5.1.1 System Tests

Actual or simulated L9

Supp. 2

4.5.1.1.1 High Pressure Injection System

a. Every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the High Pressure Injection System for emergency core cooling operation. SR 3.5.2.4 SR 3.5.2.5

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel. LA3

Supp. 2

4.5.1.1.2 Low Pressure Injection System <See 3.5.3 & 3.7>

a. Every 18 months, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to demonstrate actuation of the Low Pressure Injection System for emergency core cooling operation.

(2) Verification of the engineered safety features function of the Low Pressure Service Water pumps and manual alignment from the control room of valves LPSW-4 and LPSW-5 shall be made to demonstrate operability of the Low Pressure Injection coolers.[1] <See 3.7>

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the ES actuation signal properly; all appropriate ES actuated pump breakers shall have opened or closed, and all ES actuated valves shall have completed their travel. In addition, valves LPSW-4 and LPSW-5 shall have completed their travel. <See 3.5.3> <See 3.7>

Supp. 2

[1] The ES function of valves LPSW-4 and LPSW-5 shall be verified every 18 months. This surveillance requirement may be discontinued and replaced by the valve surveillance in 4.5.1.1.2.a.(2) when the ES signals are removed from LPSW-4 and LPSW-5. Removal of the ES signal from valves LPSW-4 and LPSW-5 is scheduled in the U3EOC16, U1EOC17, and U2EOC16 refueling outages successively.

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

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Specification 3.5.2

Supp. 1

Supp. 2

4.5.1.1.3 Core Flooding System (See 3.5.1)

a. Every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During pressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests (See 3.5.3)

4.5.1.2.1 Valves - Power Operated

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

A13

Supp. 2

b. Every 18 months, the following LPI system valves shall be cycled ~~manually~~ to verify the manual operability of these power operated valves:

SR 3.5.2.7

(1) LPI pump discharge (ES) LP-17,-18
(2) LPI discharge throttling LP-12,-14
(3) LPI discharge header crossover LP-9,-10
(4) LPI discharge to HPI/RBS ~~LP-15,-16~~ LA6

4.5.1.2.2 Check Valves (See 3.4)

Periodic individual leakage testing[a] of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakage rates and limiting conditions for operation, see Technical Specification 3.1.6.10.

Bases

The Emergency Core Cooling Systems are the principle reactor safety features in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. The HPI system test required by Specification 4.5.1.1.1 verifies that the HPI system responds as required to actuation of ES channels 1 and 2.

A2

(See 3.4)

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

Specification 3.5.3

(A1) <except as marked>

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency	
1. Control Rod Movement (1)	Movement of Each Rod	Monthly	<SEE 3.1>
Supp 2 2. Pressurizer Safety Valves	Setpoint	18 months(4)	<SEE 3.4>
3. Main Steam Safety Valves	Setpoint	18 months(4)	<SEE 3.7>
4. Refueling System Interlocks(5)	Functional	Prior to Refueling	<SEE 3.3>
5. Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly	<SEE 3.7>
6. Reactor Coolant System (2) Leakage	Evaluate	Daily	<SEE 3.4>
Supp 2 7. Emergency Condenser (6) Circulating Water System Test	Functional	18 months	<SEE 3.7>
8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	
9. Spent Fuel Cooling System	Functional	Prior to Refueling	
10. High Pressure and Low (3) Pressure Injection System <SEE 3.5.2>	Vent Pump Casings	31 days Monthly and Prior to Testing	L10
Supp 2 11. Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	<SEE 3.7>
12. MSLB Feedwater Isolation (7) Feature	Functional	18 months	
13. Essential Siphon Vacuum (8) System Test	Functional	Quarterly	

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.5.3

(A1) (except as marked)

(1) Applicable only when the reactor is critical. – (SEE 3.1 + 3.7)

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. (SEE 3.4)

Supp 1 | SR 3.5.3.2 Note

(3) Operating pumps excluded.

Supp 2 |

(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. (SEE 3.4+3.7)

(5) Applicable only to the interlocks associated with the Reactor Building Purge System. (SEE 3.3)

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature.

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit. (SEE 3.7)

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.5.3

Add SR 3.5.3.1 — M6

Add SR 3.5.3.3 — A8

Add SR 3.5.3.6 — M7

4.5 EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Core Cooling Systems

Applicability

Applies to periodic testing requirements for the Emergency Core Cooling Systems.

Objective

To verify that the Emergency Core Cooling Systems are operable.

See 3.5.2

Specification

4.5.1.1 System Tests

4.5.1.1.1 High Pressure Injection System

a. Every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the High Pressure Injection System for emergency core cooling operation.

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.

4.5.1.1.2 Low Pressure Injection System

Supp. 2

a. Every 18 months, a system test shall be conducted to demonstrate that the system is operable. ~~The test shall be performed in accordance with the procedure summarized below:~~

SR 3.5.3.4
SR 3.5.3.5

(1) A test ~~signal~~ actuation (actual or simulated) will be applied to demonstrate actuation of the Low Pressure Injection System for emergency core cooling operation. (L9)

LA3

(2) Verification of the engineered safety features function of the Low Pressure Service Water pumps and manual alignment from the control room of valves LPSW-4 and LPSW-5 shall be made to demonstrate operability of the Low Pressure Injection coolers.[1]

See 3.7

b. ~~The test will be considered satisfactory if control board indication verifies that all components have responded to the ES actuation signal properly; all appropriate ES actuated pump breakers shall have opened or closed, and all ES actuated valves shall have completed their travel.~~ In addition, valves LPSW-4 and LPSW-5 shall have completed their travel.

Supp. 2

[1] The ES function of valves LPSW-4 and LPSW-5 shall be verified every 18 months. This surveillance requirement may be discontinued and replaced by the valve surveillance in 4.5.1.1.2.a.(2) when the ES signals are removed from LPSW-4 and LPSW-5. Removal of the ES signal from valves LPSW-4 and LPSW-5 is scheduled in the U3EOC16, U1EOC17, and U2EOC16 refueling outages successively.

~~Amendment No. 228 (Unit 1)~~
~~Amendment No. 229 (Unit 2)~~
~~Amendment No. 225 (Unit 3)~~

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ITS Section 3.5

ID 100

Subject: Strengthened JFD 9 for Section 3.5 based on response made to NRC comment #2 for that section and revised DOC M2 accordingly.

TECHNICAL CHANGE - MORE RESTRICTIVE

M1 CTS 3.3.3 identifies specific values for CFT inventory and nitrogen gas pressure that must be met and requires the isolation valve be open and breakers open but does not provide periodic verification of compliance with these requirements. ITS SR 3.5.1.1 requires that each CFT valve be verified fully open every 12 hours. This ensures the CFTs are available for injection and ensures timely discovery if a valve should be less than fully open. ITS SR 3.5.1.2 and ITS SR 3.5.1.3 verify that each CFT's borated water volume and nitrogen cover pressure is sufficient to ensure adequate injection during a LOCA. SR 3.5.1.5 requires verification on a Frequency of 31 days that power is removed from each CFT isolation valve operator. The SR provides assurance that the power is removed from the valve operator and that the valve is not susceptible to an active failure. These SRs provide for timely recognition of out-of-specification conditions. The addition of the SRs constitute more restrictive requirements than those presently imposed by the CTS. The changes are consistent with the NUREG.

M2 CTS Table 4.1-3, Item 3 requires CFT boron concentration to be sampled monthly and after each makeup. ITS SR 3.5.1.4 requires CFT boron concentration be sampled every 31 days and once within 12 hours after each solution volume increase ≥ 80 gallons that is not the result of addition from a borated water source that meets CFT boron concentration requirements. The CTS frequency of monthly is equivalent to the ITS frequency of 31 days. However, the second ITS sampling frequency is more restrictive than the CTS sampling frequency of "after each makeup" since sampling will be required "once within 12 hours after each solution volume increase ...." CTS does not specify a time limit for sampling. This Completion Time is based on the need to clearly establish when the required sampling must be completed while taking into consideration the time necessary to mix the tank by convection, obtain the sample and perform the analysis. The proposed change is consistent with the NUREG.

M3 CTS 3.3.3 requires the minimum boron concentration to be within the limit specified in the COLR. The COLR does not specify an upper value for allowable boron concentration in the CFTs. ITS SR 3.5.1.4 requires that the boron concentration be within the limit specified in the COLR, which will include a maximum boron concentration limit for the CFTs. The addition of the upper limit is an appropriate more restrictive limit on operation since it ensures that boric acid precipitation will not occur in the reactor vessel following a LOCA. The addition of this upper limit is consistent with the NUREG.

M4 CTS 3.3.4 requires the BWST to be OPERABLE when the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical. ITS 3.5.4 requires the BWST to be OPERABLE during MODES 1, 2, 3, and 4. MODE 4 is defined as subcritical with the average coolant temperature > 200°F and < 250°F. CTS criteria specified as 250°F is considered more

8 NUREG SR 3.5.1.4 and SR 3.5.4.3 provide specific acceptance criteria for Core Flood Tank (CFT) and Borated Water Storage Tank (BWST) boron concentration within the SR. The NUREG is modified to refer to the boron concentration limits specified in the COLR consistent with the CTS.

9 NUREG SR 3.5.1.4 second Frequency is modified to reflect unit specific system characteristics. The normal source of makeup for the CFT is the
2 boric acid mix tank. However, there are several sources of makeup that
2 meet boron concentration requirements that can be used. Therefore, reference to the source of inventory is changed from the "borated water storage tank" to "a source that meets CFT boron concentration requirements." Inventory makeup to the CFTs is sampled to demonstrate an acceptable boron concentration of the makeup water prior to its admittance into the CFT. Non-sampled makeup or makeup from other non-verified sources will continue to require the initiation of sampling in accordance with the intent of the SR 3.5.1.4 second Frequency criteria. The NUREG SR 3.5.1.4 second Frequency is also modified to extend the
2 NUREG Completion Time of 6 hours to 12 hours. ONS design does not
2 provide a method of mixing (e.g., recirculation) after CFT makeup and
2 relies on convection mixing. The additional 6 hours is needed to allow
2 time for convection mixing following CFT makeup, then obtain the sample and perform the sample analysis.

10 NUREG SR 3.5.1.5 is modified to remove reference to RCS pressure as a condition of Applicability for the SR. The NUREG allowance is based on inadvertent closure being prevented by an RCS pressure interlock associated with these valves. These interlocks do not exist for the ONS CFT outlet valves. These valves are operator controlled. CTS 3.3.3 requires that the CFT discharge valve breaker be "locked open and tagged" as a condition of OPERABILITY for the CFT. The removal of the pressure requirement in SR 3.5.1.5 makes the SR consistent with the Applicability of LCO 3.5.1 which imposes a more restrictive pressure requirement of 800 psig.

11 NUREG LCO 3.5.2 Applicability is modified to specify that it is applicable in MODES 1 and 2 and in MODE 3 with Reactor Coolant System (RCS) temperature greater than 350°F. This Applicability is consistent with the high pressure injection (HPI) requirements of CTS Specification 3.3.2.

In addition, the Note modifying LCO 3.5.2 is deleted because HPI OPERABILITY is not required until RCS temperature exceeds 350°F. LTOP requirements (ITS LCO 3.4.12) are imposed when RCS temperature is less than 325°F. Therefore, there is no overlap of Applicability between these two Specifications and the Note is not required.

NUREG 3.5.2 Required Action B.2 (ITS 3.5.2 Required Action F.2) is modified to establish consistency between this Required Action and the

LCO 3.5.2 Applicability. This change maintains consistency with the Application and Usage of MODES and Applicability.

12 NUREG LCO 3.5.2 and 3.5.3 are modified to incorporate ONS CTS requirements necessary to appropriately consider unique design and analysis requirements. ITS LCO 3.5.2 addresses only HPI System requirements. The ONS HPI System consists of two trains with three HPI pumps that can feed either train. At power levels > 75%, three pumps and the HPI crossover valves are required to be OPERABLE and the suction header is required to be cross-connected to meet safety analyses assumptions. When less than three pumps are OPERABLE, the HPI pump discharge headers must be hydraulically separated between trains to meet safety analyses assumptions.

ITS 3.5.2, ACTIONS A through E are added to appropriately consider unique design and analysis requirements. Proposed Condition A addresses the situation where the required HPI pump or HPI crossover valve(s) are inoperable or the suction header is not cross-connected when reactor power level is > 75% power. Required Action A.1 requires restoration in 72 hours. If not restored, ACTION B then requires a power reduction to ≤ 75% within 12 hours (Required Action B.1). Condition C addresses the situation where one HPI train is inoperable. Required Action C.2 requires restoration within 72 hours. Required Action C.1 addresses the situation where an HPI train is inoperable and not capable of being manually realigned when the power level is > 75%. C.1 requires that THERMAL POWER be reduced to ≤ 75% RTP within 3 hours. This action is necessary because one train of HPI is not capable of meeting safety analysis requirements for power levels in excess of 75% power. Condition D addresses the situation where one LPI-HPI flow path is inoperable. Required Action D.1 allows 72 hours to restore the flow path. Condition E addresses the situation where less than three HPI pumps are OPERABLE. Required Action E.1 allows 72 hours to hydraulically separate the discharge header.

NUREG SRs 3.5.2.2, 3.5.2.4,, 3.5.2.5, 3.5.2.6 and 3.5.2.9 are modified to address only HPI requirements. ITS SR 3.5.2.7 is added to demonstrate the ability to manually transfer HPI suction from the BWST to the LPI discharge by manually cycling the LPI discharge valves to the LPI-HPI flow path.

13 NUREG 3.5.2 Condition B is modified to add a second and third entry condition that is premised on the inoperability two HPI trains or two LPI-HPI flow paths. The change retains CTS 3.3.1 Condition F provisions. In this case, the safety function provided by the HPI is not capable of being satisfied and the unit is operating in a condition outside of the accident analyses. This condition would ordinarily require entry into LCO 3.0.3. However, LCO 3.0.3 does not provide an explicit Completion Time for placing the unit in MODE 3 with RCS temperature less than or equal to 350°F. By providing this explicit entry condition, Proposed LCO 3.5.2 Required Actions E.1 and E.2 direct

the appropriate remedial measures and explicitly establish the Completion Times for the Required Actions.

14 NUREG SR 3.5.2.1, SR 3.5.2.7 and SR 3.5.2.8 are not included in the proposed ONS ITS since there are no comparable ONS CTS requirements.

NUREG SR 3.5.2.1 provides assurance that valves cannot change position as the result of an active failure. SR 3.5.2.1 is not included since the ONS ECCS trains contain no power operated valves that are required to remain de-energized or whose control circuits require key locking the control in the correct position. Information Notice 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRS" was reviewed for applicability. This information notice addresses valve misalignments during testing that could result in both trains of ECCS being rendered inoperable. ONS is not susceptible to the events described in the information notice.

NUREG SR 3.5.2.7 is not included since the HPI trains do not contain stop check valves whose purpose is to balance flow or prevent HPI pump runout. ONS uses flow orifices which are not susceptible to repositioning.

NUREG SR 3.5.2.8 is not included since ONS LPI trains do not contain automatic flow controlling throttle valves. Flow is controlled manually using installed flow indicators which are required OPERABLE by LCO 3.3.8, "Post Accident Monitoring Instrumentation."

15 NUREG LCO 3.5.2 and 3.5.3 are modified to incorporate ONS CTS requirements necessary to appropriately consider unique design and analysis requirements. ITS LCO 3.5.3 addresses only LPI System requirements. NUREG LCO 3.5.3 is modified to require two LPI trains to be OPERABLE in MODES 1, 2 and 3 and one LPI train to be OPERABLE in MODE 4 (as indicated by ITS LCO 3.5.3 Note 1).

NUREG LCO 3.5.3 Note 2, as renumbered by TSTF-090, Revision 1, is deleted since ITS 3.5.3 applies to LPI only. LCO 3.5.3 Note 1, as added by TSTF-090, Revision 1, is modified (and renumbered as ITS LCO 3.5.3 Note 2) to apply "during alignment, when aligned and when operating" to remove the confusion associated with whether the DHR/LPI train was OPERABLE during the swapover and whether the DHR/LPI pump had to be running in order to satisfy the requirement that the system be in operation. The annotation that the manual control can be accomplished either locally or remotely preserves current operational flexibility.

Appropriate SRs, similar to those adopted for the HPI System, are added for the LPI System (ITS SRs 3.5.3.1 - SR 3.5.3.6) to demonstrate compliance with LCO requirements.

16 NUREG LCO 3.5.3 Actions were altered, while retaining the original intent of the Required Actions, in order to properly reflect the

ITS Section 3.5

ID 102

Subject: Revised ITS 3.5.4.1 to adopt NUREG note.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
2 2 2 2	SR 3.5.4.1	----------NOTE---------- Only required to be performed when ambient air temperature is < 45°F or > 115°F. ---------- Verify BWST borated water temperature is ≥ 45°F and ≤ 115°F.	24 hours
	SR 3.5.4.2	Verify BWST borated water volume is ≥ 350,000 gallons.	7 days
	SR 3.5.4.3	Verify BWST boron concentration is within the limit specified in the COLR.	7 days

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification every 24 hours that the BWST water temperature is within the specified temperature band ensures that the fluid will not freeze and that the fluid temperature entering the reactor vessel will not be colder than assumed in the reactor vessel stress analysis; and the fluid temperature entering the reactor vessel will not be hotter than assumed in the LOCA analysis. The 24 hour Frequency is sufficient to identify a temperature change that would approach either temperature limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that requires the Surveillance to be performed only when ambient air temperatures are outside the operating temperature limits of the BWST. With ambient temperature within this band, the BWST temperature should not exceed the limits.

SR 3.5.4.2

Verification every 7 days that the BWST contained volume is $\geq$ 350,000 gallons (46.0 ft.) ensures that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. Since the BWST volume is normally stable, a 7 day Frequency has been shown to be appropriate through operating experience.

SR 3.5.4.3

Verification every 7 days that the boron concentration of the BWST fluid is within the required band ensures that the reactor will remain subcritical following a LOCA. Since the BWST volume is normally stable, a 7 day sampling Frequency is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. 10 CFR 50.36.

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.5.4.1	----------NOTE---------- Only required to be performed when ambient air temperature is < [40] °F or > [100] °F. ----------		DOC M5 DOC L8
	Verify BWST borated water temperature is ≥ ~~[40]~~ 45 °F and ≤ ~~[100]~~ 115 °F.	24 hours	
SR 3.5.4.2	Verify BWST borated water volume is ≥ ~~[415,200 gallons, [ ] ft and~~ 350,000 gallons and ≤ 449,000 gallons, [ ] ft].	7 days	DOC M5 DOC L8
SR 3.5.4.3	Verify BWST boron concentration is ≥ [2270] ppm and ≤ [2450] ppm.	7 days	T4.1-3, Item 2

Supp 2

(45) (115) (6) (6) (7) (8)

within the limit specified in the COLR

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification every 24 hours that the BWST water temperature is within the specified temperature band ensures that the ~~boron will not precipitate;~~ and that the fluid will not freeze, the fluid temperature entering the reactor vessel will not be colder than assumed in the reactor vessel stress analysis; and the fluid temperature entering the reactor vessel will not be hotter than assumed in the LOCA analysis. The 24 hour Frequency is sufficient to identify a temperature change that would approach either temperature limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that requires the Surveillance to be performed only when ambient air temperatures are outside the operating temperature limits of the BWST. With ambient temperatures within this band, the BWST temperature should not exceed the limits.

SR 3.5.4.2

Verification every 7 days that the BWST contained volume is ~~within the required range~~ ≥ 350,000 gallons (46.0 ft) ensures that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. Since the BWST volume is normally stable ~~and provided with a low level alarm~~, a 7 day Frequency has been shown to be appropriate through operating experience.

SR 3.5.4.3

Verification every 7 days that the boron concentration of the BWST fluid is within the required band ensures that the reactor will remain subcritical following a LOCA. Since the BWST volume is normally stable, a 7 day sampling Frequency is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. ~~FSAR, Section [6.1].~~ 10 CFR 50.36

corrective actions should the LCO not be met. NUREG Condition B is designated as ITS Condition A. Condition A is entered when one train of LPI is inoperable in MODES 1, 2 or 3. ITS Required Action A.1 allows 72 hours to restore the LPI train to OPERABLE status. This is consistent with the CTS 3.3.2.a(2) restoration time. The 72 hour Completion Time is an acceptable allowance based on the fact that the redundant LPI train can still satisfy the required ECCS safety function for the specified LCO Applicability. Condition C is entered when the Required Action and associated Completion Time of Condition A are not met. ITS Required Action C.1 requires that the unit be in MODE 3 within 12 hours and MODE 4 within 60 hours. This Completion Time in conjunction with the Completion Time of ITS Required Action A.1 (72 hours) is in accordance with CTS 3.3.2(a) requirements for the restoration of operability or completion of compensatory measures for the LPI systems. Further, the combination of ITS Conditions A and C preserves the philosophy of removing the unit from the MODES or other specified conditions for Applicability.

NUREG Condition A is designated as ITS Condition E. Condition E is entered when the required LPI train is inoperable during MODE 4. ITS Required Action E.1 requires that action be immediately initiated to restore the decay heat removal (DHR) loop to an OPERABLE status. This Required Action and its associated Completion Time are premised on the recognition that an ECCS safety function has been lost. Further, this Required Action and its associated Completion Time are structured such that no requirement for a reduction in RCS temperature exists (i.e., LCO 3.0.3 is not entered). If both LPI trains are inoperable, the corrective action is to restore at least one LPI train to an OPERABLE status prior to cooling the unit down and into a MODE that requires operation of the DHR mode of the LPI System. Required Action E.2 is inserted to provide a Required Action to place the unit in MODE 5 if the DHR mode of one LPI train is available despite the inoperability of both of the LPI trains. ONS has a third LPI pump (non ES) that can be used for DHR. This Required Action is conditional based on a NOTE that directs that this action is required only if the DHR mode of one LPI train is OPERABLE. If the cause of the inoperability for both LPI trains also made the DHR mode inoperable, then no attempt to cool down the unit is required. Required Action E.2 is inserted to ensure that a cooldown to MODE 5 is initiated provided the required DHR capability exists. These changes are consistent with NUREG LCO 3.4.5 and LCO 3.4.6 Actions when a decay heat removal system is unavailable.

2 17 Not used.

18 In general, current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown when an LCO is not met (refer to JFD 5). However, the CTS shutdown requirement for the BWST not being restored to OPERABLE status within one hour is to place the unit in Hot Shutdown (equivalent to ITS MODE 3) within 6 hours and Cold Shutdown (equivalent to ITS MODE 5) within and additional 30 hours, when the concentrated boric acid

tank is available. When the CBAST is not available, no time is allowed and CTS LCO 3.0 must be entered requiring the unit to be placed in Hot Shutdown in 12 hours and in Cold Shutdown within the following 24 hours. To maintain consistency with CTS LCO 3.0, the Completion Time for ITS 3.5.4 Required Action C.1 is changed to 12 hours.

19 NUREG SR 3.5.2.3 is modified to adopt the current method of minimizing the potential for water hammer and pump cavitation. NUREG SR 3.5.2.3 verifies that the ECCS piping is full of water. At ONS, the physical design of the HPI and LPI systems are such that the SR could not be applied to all portions of the piping because of the inability to perform venting operations to satisfy the SR due to the absence of vents, physical danger associated with the evolution or due to localized radiation levels. Therefore, the ITS requires HPI/LPI pump casings be vented every 31 days consistent with CTS Table 4.1-2, Item 10 requirements. Venting the HPI/LPI pump casings periodically reduces the potential that such voids and pockets of entrained gases can adversely affect operation of the HPI/LPI Systems.

20 Not used.

21 NUREG 3.5.3 is modified to incorporate ONS CTS requirements necessary to appropriately consider unique analysis requirements. In the event of a Core Flood line break (classified as a small break LOCA) concurrent with a single failure on the unaffected LPI train opposite the Core Flood line break, the LPI discharge header crossover valves must be capable of being manually opened. This action, along with manually opening associated LPI cooler outlet throttle valve and LPI header isolation valve will allow cross-connection of the OPERABLE LPI pump discharge to the intact LPI/Core Flood tank header to provide abundant long term cooling. ITS 3.5.3 LCO Note 3 is added to explicitly require the LPI discharge header crossover valves to be OPERABLE to manually open in MODES 1, 2, and 3. ITS 3.5.3 ACTION B requires the LPI discharge header crossover valves to be restored to OPERABLE status with 72 hours of being discovered manually inoperable to open in MODE 1, 2, and 3. ITS 3.5.3 ACTION D requires LCO 3.0.3 be entered immediately when one LPI train is inoperable in MODES 1, 2, and 3 concurrent with the LPI discharge header crossover valves are inoperable to open manually in MODE 1, 2, and 3. The 72 hour Completion time of ITS 3.5.3 Required Action B.1 is based on NRC recommendations (NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975) that are based on risk evaluation and are reasonable time for repairs. This is considered appropriate since loss of crossover capability is equivalent to a loss of a single train of LPI as it relates to providing abundant long term cooling. ACTION D is added to preclude concurrent inoperability of an LPI train and the LPI discharge header crossover valves since this could result in the inability to provide abundant long term cooling to the reactor. ITS SR 3.5.3.7 is added to require each LPI discharge header crossover valve, LPI cooler outlet throttle valve and LPI header isolation valve to be

cycled open once every 18 months. This is consistent with CTS requirements for these valves and is considered appropriate based on operating experience.

ITS Section 3.5

ID 132

Subject: Revised 3.5.2 to add missing CTs for RA's A.2 and A.3, renumbered the 2nd SR 3.5.2.1 to be SR 3.5.2.2.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 High Pressure Injection (HPI)

LCO 3.5.2 Two HPI trains and two LPI-HPI flow paths shall be OPERABLE.

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

1. When THERMAL POWER is > 75% RTP, three HPI pumps and the HPI crossover valves shall be OPERABLE and the suction header shall be cross-connected.

2. When less than three HPI pumps are OPERABLE, the HPI pump discharge header shall be hydraulically separated between trains.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required HPI pump inoperable. OR One or more required HPI crossover valve(s) inoperable. OR HPI suction header not cross-connected when required.	A.1 Restore required HPI pump to OPERABLE status. AND A.2 Restore required HPI crossover valve(s) to OPERABLE status. AND A.3 Cross-connect HPI suction header.	72 hours 72 hours 72 hours

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
	SR 3.5.2.1	Verify each HPI manual and non-automatic power operated valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
2	SR 3.5.2.2	----------NOTE---------- Not applicable to operating HPI pump(s). ---------- Vent each HPI pump casing.	31 days
	SR 3.5.2.3	Verify each HPI pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
	SR 3.5.2.4	Verify each HPI automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
	SR 3.5.2.5	Verify each HPI pump starts automatically on an actual or simulated actuation signal.	18 months

(continued)

INSERT 3.5-4B

	A.	One required HPI pump inoperable. OR One or more required	A.1	Restore required HPI pump to OPERABLE status. AND	72 hours
2		HPI crossover valve(s) inoperable. OR HPI suction header	A.2	Restore required HPI crossover valve(s) to OPERABLE status. AND	72 hours
2		not cross-connected when required.	A.3	Cross-connect HPI suction header.	72 hours
	B.	Required Action and associated Completion Time of Condition A not met.	B.1	Reduce THERMAL POWER to ≤ 75% RTP.	12 hours
	C.	One HPI train inoperable.	C.1	--------NOTE-------- Applicable when the train is incapable of automatic actuation and is also incapable of actuation through manual alignment. ---------- Reduce THERMAL POWER to ≤ 75% RTP. AND	3 hours
			C.2	Restore train to OPERABLE status.	72 hours

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Specification 3.6

4.4.3 Containment Hydrogen Control Systems

LA3

Applicability

Applies to the Containment Hydrogen Control Systems.

Objective

To verify that the Containment Hydrogen Control Systems are operable.

Specifications

Supp. 2

4.4.3.1 Containment Hydrogen Control System Piping

Every 18 months, the permanent piping for the Containment Hydrogen Control System shall be tested as follows:

a. The post-LOCA flow paths shall be verified by connecting and operating either the Hydrogen Purge Unit or the Hydrogen Recombiner through each flow path as follows:

1. The hydrogen Recombiner flow path circulates Reactor Building atmosphere at a flow greater than 50 SCFM.

2. The Hydrogen Purge flow path removes Reactor Building atmosphere and discharges to the Unit vent stack at a flow greater than or equal to 45 SCFM.

b. The blind isolation flanges on the Containment Hydrogen Control System permanent piping shall be leak tested after each installation to ensure adequate isolation.

4.4.3.2 Containment Hydrogen Recombiner System Operational Performance Testing

Supp. 2

a. The testing requirement of this section may be performed without connecting the system to either of the Reactor Buildings.

b. Every 18 months:

1. Visual inspection of the unit.

2. Calibrate all recombiner instrumentation and control circuits.

3. Operate a recombiner unit at design flow rate 10% and allow unit to reach recombination temperature.

4.4.3.3 Reactor Building Hydrogen Purge System, Pre-Operational Testing

a. Prior to declaring this system operable, a Pre-operational system test shall be performed.

3 of 4

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Specification 3.6.3

(A1) (except as marked)

4.4.4 Reactor Building Purge System

Applicability

Applies to the Reactor Building Purge System.

Objective

To verify that the Reactor Building Purge System is operable.

Specification

(LA2)

4.4.4.1 Each shutdown, when the purge valves have been operated, leakage integrity tests shall be performed on the containment purge isolation valves after final closing and prior to going above hot shutdown. If the purge valves have not been operated, leakage integrity tests shall be performed prior to going above hot shutdown unless such tests have been conducted within the proceeding six months. If the acceptance criteria of Specification 4.4.1.2.3 are not met, Specification 3.6.6 shall apply. Unit shutdown to conduct the test and/or effect repairs is specifically not required.

SR 3.6.3.1

4.4.4.2 Monthly, when the unit is above ~~250°F and 350 psig~~ MODES 3, the containment purge isolation valves shall be verified closed. (sealed) (M13)

(LA8)

4.4.4.3 Every ~~18~~ months, the valve seals of the containment purge isolation valves shall be visually inspected and adjusted ~~or replaced~~ as appropriate.

(M7)

4.4.4.4 Prior to use of the purge system at conditions between cold shutdown and 250°F and 350 psig, the isolation valves shall be exercise tested in accordance with the requirements (except test frequency) of the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI. (see 3.9)

4.4.4.5 The pneumatically operated purge isolation valves shall be verified to close in response to a control signal from RIA-45 when the system is tested prior to refueling operations per Specification 3.8.10.

Bases

Leakage integrity tests of the purge supply and isolation valves are conducted in order to identify excessive degradation of the resilient seals. Excessive leakage past resilient seals is typically caused by severe environmental conditions and/or wear due to frequent use.

The pneumatically operated purge isolation valves are tested prior to refueling operations because the only automatic isolation system in service at refueling is through RIA-45, which only closes the pneumatic isolation valves.

(A2)

5 of 5

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Specification 3.6.5

A1 <except as marked>

4.5.2 Reactor Building Cooling Systems

Applicability

Applies to testing of the Reactor Building Cooling Systems.

Objective

To verify that the Reactor Building Cooling Systems are operable.

Specification

L8 Actual or simulated

Add SR 3.6.5.1 M18

Add SR 3.6.5.3 A14

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

Supp. 2

a. (1) Once per Every 18 months, a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the Reactor Building Spray System.

SR 3.6.5.5
SR 3.6.5.6

(2) The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly, the appropriate pump breakers shall have closed, and all valves shall have completed their travel. LA1

b. Station compressed air will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every ten years. LA1

SR 3.6.5.8

4.5.2.1.2 Reactor Building Cooling System

Supp. 2

SR 3.6.5.7

a. Once per Every 18 months[1], a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below: Actual or Simulated L8

(1) A test signal will be applied to actuate the Reactor Building Cooling System for reactor building cooling operation.

(2) Verification of the engineered safety features function of the Low Pressure Service Water System which supplies coolant to the reactor building coolers shall be made to demonstrate operability of the coolers. <See 3.7>

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate valves have completed their travel, and fans are running at half speed. LA1

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Supp. 2

[1]A one-time extension of the Reactor Building Cooling system test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16. A36

Specification 3.6.5

A1 Except as Marked

4.5.3 Containment Heat Removal Capability

Applicability

Applies to verification of adequate containment heat removal capability.

Objective

To verify that containment heat removal capability is sufficient to maintain post accident conditions within design limits.

Specification

Suppl. 2 SR 3.6.5.4

4.5.3.1 Containment Heat Removal Capability

a. Every 18 months, containment heat removal capability shall be verified to be sufficient to maintain post accident conditions within design limits.

b. In addition to the requirements of 4.5.3.1.a, on a frequency consistent with the LPI cooler and RBCU fouling rate, containment heat removal capability shall be verified to be sufficient to maintain post accident conditions within design limits.

LA5

Bases

The safety functions of the LPI system, RB Spray system, and RBCUs include maintaining containment pressure and temperature below design limits following an accident. This surveillance assures that containment heat removal capability is adequate assuming a worst-case single failure. Specification 4.5.3.1.a requires that at a minimum the surveillance be performed every 18 months. In addition, since service induced fouling can reduce containment heat removal capability, Specification 4.5.3.1.b requires that a fouling rate be determined in order to establish a more frequent test interval if required.

REFERENCES:

FSAR Section 6.2
FSAR Section 15.14

A2

Add SR 3.6.5.2 — M15

5 of 5

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

ITS Section 3.6 3.7 4.0 5.0

ID 139

Subject: Revised ITS to remove Penetration Room Ventilation System Tech Spec change 95-04

Specification 3.6.1

(A1) <except as marked>

4.4 REACTOR BUILDING

4.4.1 Containment Leakage Tests

Applicability

Applies to Containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

(A9) in accordance with the Containment Leakage RATE Testing Program

~~4.4.1.1 Integrated Leak Rate Tests~~

SR3.6.1.1

The containment leakage rate shall be determined as required by 10CFR50.54 (o) and 10CFR50, Appendix J, Option B, including any approved exemptions, using the guidelines of Regulatory Guide 1.163, dated September, 1995.

4.4.1.1.1 Acceptance Criteria

Sull. 2

The overall acceptance containment leakage rate is determined by the preoperational leakage rate test and shall not exceed L_a, 0.25 weight percent of containment air per 24 hours at P_a, .59 psig. Any leakage in excess of 50% of the total allowed containment leakage shall be demonstrated to be to the penetration room. Containment leakage prior to startup following a Type A test shall not exceed .75 L_a.

<See 5.0>

~~4.4.1.2 Local Leak Rate Testing~~

~~4.4.1.2.1 Scope of Testing~~

SR3.6.1.2

The local leak rate shall be measured for the containment penetrations in accordance with the criteria specified in Appendix J of 10CFR50, Option A.

(A19) Add acceptance criteria of SR3.6.1.2

4.4.1.2.2 Frequency of Test

Local leak rate tests shall be conducted with gas at a pressure of not less than 59 psig during each reactor shutdown for refueling or other convenient interval but in no case at intervals greater than 24 months.

<See 5.0>

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Amendment No. 218 (Unit 1)
Amendment No. 218 (Unit 2)
Amendment No. 215 (Unit 3)

Specification 3.6.2

A17

<See 5.0>

4.4.1.2.3 Acceptance Criteria

Supp. 2

The combined leakage rate from all penetrations and isolation valves shall not exceed 0.125 weight percent of the postulated post-accident containment air mass per 24 hours at 59 psig.

4.4.1.3 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a local leak rate test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.1 and 4.4.1.2.3, respectively.

4.4.1.4 Isolation Valve Functional Tests

Inservice testing of ASME Code Class 1, 2, and 3 valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

<See 3.6.3>

Specification 3.6.3

(A1) <except as marked>

<See 5.0>

4.4.1.2.3 Acceptance Criteria

Supp. 2

The combined leakage rate from all penetrations and isolation valves shall not exceed 0.125 weight percent of the postulated post-accident containment air mass per 24 hours at 59 psig.

4.4.1.3 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a local leak rate test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.1 and 4.4.1.2.3, respectively.

~~4.4.1.4 Isolation Valve Functional Tests~~

SR 3.6.3.4

Inservice testing of ASME Code Class 1, 2, and 3 valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

LA7

the IST Program — A15

Add SR 3.6.3.5 — M19

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Specification 3.7.1

(A1) except as marked

(A24) in accordance with the Inservice Testing Program

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency	
1.	Control Rod Movement [1]	Movement of Each Rod	Monthly	<SEE 3.1>
2.	Pressurizer Safety Valves	Setpoint	18 months[4]	<SEE 3.4>
Supp 2 / SR 3.7.1.1 3.	Main Steam Safety Valves	Setpoint (each required)	18 months[4]	
4.	Refueling System Interlocks[5]	Functional	Prior to Refueling	<SEE 3.3>
5.	Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly	<SEE 3.7.2>
6.	Reactor Coolant System [2] Leakage	Evaluate	Daily	<SEE 3.4>
Supp 2 7.	Emergency Condenser [9] Circulating Water System Test	Functional	18 months	<SEE 3.7.8>
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	R3
9.	Spent Fuel Cooling System	Functional	Prior to Refueling	R2
10.	High Pressure and Low [3] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	<SEE 3.5>
Supp 2 11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	<SEE 3.7.5>
12.	MSLB Feedwater Isolation [7] Feature	Functional	18 months	<SEE 3.7.3>
13.	Essential Siphon Vacuum [8] System Test	Functional	Quarterly	<SEE 3.7.8>

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.7.1

(1) Applicable only when the reactor is critical. <SEE 3.1 + 3.7.2>

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. <SEE 3.4>

(3) Operating pumps excluded. <SEE 3.5>

Supp 2

(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. Inservice Testing Program A24 LA2

(5) Applicable only to the interlocks associated with the Reactor Building Purge System. <SEE 3.3>

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System. <SEE 3.7.8>

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature. <SEE 3.7.3>

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit. <SEE 3.7.8>

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.7.2

(A1) except as marked

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency	
1.	Control Rod Movement [1]	Movement of Each Rod	Monthly	<SEE 3.1>
2.	Pressurizer Safety Valves	Setpoint	18 months[4]	<SEE 3.4>
3.	Main Steam Safety Valves	Setpoint	18 months[4]	<SEE 3.7.1>
4.	Refueling System Interlocks[5]	Functional	Prior to Refueling	<SEE 3.3>
5.	Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly	(A3)
6.	Reactor Coolant System [2] Leakage	Evaluate	Daily	<SEE 3.4>
7.	Emergency Condenser [6] Circulating Water System Test	Functional	18 months	<SEE 3.7.8>
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	<SEE 3.7.1>
9.	Spent Fuel Cooling System	Functional	Prior to Refueling	<SEE 3.7.1>
10.	High Pressure and Low [3] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	<SEE 3.5>
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	<SEE 3.7.5>
12.	MSLB Feedwater Isolation [7] Feature	Functional	18 months	<SEE 3.7.3>
13.	Essential Siphon Vacuum [8] System Test	Functional	Quarterly	<SEE 3.7.8>

Supp 2

Supp 2

Supp 2

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.7.2

M3

MODE 1
MODES 2+3 except when all TSVs are closed

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. <SEE 3.4>

(3) Operating pumps excluded. <SEE 3.5>

Supp 2

(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. <SEE 3.7.1 + 3.4>

(5) Applicable only to the interlocks associated with the Reactor Building Purge System. <SEE 3.3>

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System. <SEE 3.7.8>

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature. <SEE 3.7.3>

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit. <SEE 3.7.8>

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

TSC 95-03

Pack 3

Specification 3.7.L

A1 Except as marked

4.8 MAIN STEAM STOP VALVES

Applicability

Applies to the main steam stop valves.

Objective

To verify the ability of the main steam stop valves to close upon signal.

A25

Specification

Turbine

Suppl. 2 SR3.7.2.1 SR3.7.2.2

4.8 Using Channels A and B, the operation of each of the main steam stop valves shall be tested every 18 months to demonstrate a closure time of one second or less in Channel A and a closure time of 15 seconds or less for Channel B.

Bases

The main steam stop valves limit the Reactor Coolant System cooldown rate and resultant reactivity insertion following a main steam line break accident. Their ability to promptly close upon redundant signals will be verified every 18 months. Channel A solenoid valves are designed to close all four turbine stop valves in 240 milliseconds. The backup Channel B solenoid valves are designed to close the turbine stop valves in approximately 12 seconds.

Using the maximum 15 second stop valve closing time, the fouled steam generator inventories and the minimum tripped rod worth with the maximum stuck rod worth, an analysis similar to that presented in FSAR Section 15.13, (but considering a blowdown of both steam generators) shows that the reactor will remain sub-critical after reactor trip following a double-ended steam line break.

REFERENCES

(1) FSAR, Section 10.3.4, and 15.13

A2

Add LCO
Add Actions

M1

Add Note to SR3.7.2.1
+ Note to SR3.7.2.2

A18

3 of 3

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Specification 3.7.3

(A1) except as marked

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency	
1.	Control Rod Movement [1]	Movement of Each Rod	Monthly	<SEE 3.1>
2.	Pressurizer Safety Valves	Setpoint	18 months[4]	<SEE 3.4>
3.	Main Steam Safety Valves	Setpoint	18 months[4]	<SEE 3.7.1>
4.	Refueling System Interlocks[5]	Functional	Prior to Refueling	<SEE 3.3>
5.	Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly	<SEE 3.7.2>
6.	Reactor Coolant System [2] Leakage	Evaluate	Daily	<SEE 3.4>
7.	Emergency Condenser [6] Circulating Water System Test	Functional	18 months	<SEE 3.7.8>
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	<SEE 3.7.1>
9.	Spent Fuel Cooling System	Functional	Prior to Refueling	
10.	High Pressure and Low [3] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	<SEE 3.5>
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	<SEE 3.7.5>
12.	MSLB Feedwater Isolation [7] Feature	Functional	18 months	LA14
13.	Essential Siphon Vacuum [8] System Test	Functional	Quarterly	<SEE 3.7.8>

Supp 2

Supp 2

Supp 2

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.7.3

(A1) <except as marked>

(1) Applicable only when the reactor is critical. -<SEE 3.1 + 3.7.2>

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. -<SEE 3.4>

(3) Operating pumps excluded. -<SEE 3.5>

Supp 2

(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. <SEE 3.4 + 3.7.1>

(5) Applicable only to the interlocks associated with the Reactor Building Purge System. -<SEE 3.3>

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System. -<SEE 3.7.8>

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature. LA14

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit. -<SEE 3.7.8>

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Page 3 of 3

Specification 3.7.5

A1 (except as marked)

4.9 EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine-driven and motor-driven emergency feedwater pumps and associated valves.

Objective

To verify that the emergency feedwater pumps and associated valves are operable.

Specification

LA4

SR 3.7.5.2

4.9.1 Pump Test

The turbine-driven and motor-driven feedwater pumps shall be operated on recirculation to the upper surge tank for a minimum of one hour in accordance with the requirements of Specification 4.0.4.

A30

4.9.2 Valve Test

the Inservice Testing Program

Automatic valves in the emergency feedwater flow path will be determined to be operable in accordance with the requirements of Specification 4.0.4.

LA13

Snip. 2

4.9.3 System Flow Test

LA4

Prior to Unit operation above 25% Full Power following any modifications or repairs to the emergency feedwater system which could degrade the flow path and at least once every 18 months, the emergency feedwater system shall be given either a manual or an automatic initiation signal.

1

SR 3.7.5.3

SR 3.7.5.4

4.9.4 Acceptance Criteria

LA4

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly. In addition, during operation of the System Flow Test (Item 4.9.3 above), flow to the steam generators shall be verified by control room indication.

Bases

The monthly testing frequency is sufficient to verify that the emergency feed-water pumps are operable. Verification of correct operation is made both from the control room instrumentation and direct visual observation of the pumps. The parameters which are observed are detailed in the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI. The System Flow Test verifies correct total system operation following modifications or repairs.

A2

REFERENCES

(1) FSAR, Section 10.4.7.4

4 of 7

Specification 3.7.5

(A1) except as marked

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency	
1. Control Rod Movement [1]	Movement of Each Rod	Monthly	<SEE 3.1>
2. Pressurizer Safety Valves	Setpoint	18 months[4]	<SEE 3.4>
3. Main Steam Safety Valves	Setpoint	18 months[4]	<SEE 3.7.1>
4. Refueling System Interlocks[5]	Functional	Prior to Refueling	<SEE 3.3>
5. Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly	<SEE 3.7.2>
6. Reactor Coolant System [2] Leakage	Evaluate	Daily	<SEE 3.4>
7. Emergency Condenser [6] Circulating Water System Test	Functional	18 months	<SEE 3.7.8>
8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	<SEE 3.7.1>
9. Spent Fuel Cooling System	Functional	Prior to Refueling	
10. High Pressure and Low [3] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	<SEE 3.5>
11. Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	~~Functional~~ Insert C 4.1-9A / Insert C 4.1-9B (M4)	18 months	
12. MSLB Feedwater Isolation [7] Feature	Functional	18 months	<SEE 3.7.3>
13. Essential Siphon Vacuum [8] System Test	Functional	Quarterly	<SEE 3.7.8>

Supp 2

Supp 2 SR 3.7.5.3+4

Supp 2

Supp 1

Add SR 3.7.5.1 — (M5)

Add SR 3.7.5.5 — (M16)

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.7.7

(A1 except as marked)

4.5 EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Core Cooling Systems

Applicability

Applies to periodic testing requirements for the Emergency Core Cooling Systems.

Objective

To verify that the Emergency Core Cooling Systems are operable.

Specification

4.5.1.1 System Tests (See 3.5)

Supp. 2

4.5.1.1.1 High Pressure Injection System

a. Every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the High Pressure Injection System for emergency core cooling operation.

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.

Supp. 2

4.5.1.1.2 Low Pressure Injection System (See 3.5)

a. Every 18 months, ~~a system test shall be conducted to demonstrate~~ that the system is operable. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to demonstrate actuation of the Low Pressure Injection System for emergency core cooling operation.

(M11)

SR 3.7.7.3

(2) Verification of the engineered safety features function of the Low Pressure Service Water pumps and manual alignment from the control room of valves LPSW-4 and LPSW-5 shall be made to demonstrate operability of the Low Pressure Injection coolers.[1]

(LA7)

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the ES actuation signal properly; all appropriate ES actuated pump breakers shall have opened or closed, and all ES actuated valves shall have completed their travel. In addition, valves LPSW-4 and LPSW-5 shall have completed their travel.

(See 3.5)

(LA7)

Supp. 2

SR 3.7.7.2

(M12)

[1] The ES function of valves LPSW-4 and LPSW-5 shall be verified every 18 months. This surveillance requirement may be discontinued and replaced by the valve surveillance in 4.5.1.1.2.a.(2) when the ES signals are removed from LPSW-4 and LPSW-5. Removal of the ES signal from valves LPSW-4 and LPSW-5 is scheduled in the U3EOC16, U1EOC17, and U2EOC16 refueling outages successively.

(A10)

Add SR 3.7.7.1 (M10)

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

2 of 3

Specification 3.7.7

(A1 Except as marked)

4.5.2 Reactor Building Cooling Systems

Applicability

Applies to testing of the Reactor Building Cooling Systems.

Objective

To verify that the Reactor Building Cooling Systems are operable.

Specification

4.5.2.1 System Tests (See 3.6)

Supp. 2

4.5.2.1.1 Reactor Building Spray System

a. (1) Every 18 months, a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the Reactor Building Spray System.

(2) The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly; the appropriate pump breakers shall have closed, and all valves shall have completed their travel.

b. Station compressed air will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every ten years.

Supp. 2

4.5.2.1.2 Reactor Building Cooling System

a. Every 18 months[1] a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the Reactor Building Cooling System for reactor building cooling operation.

SR 3.7.7.2 (2) Verification of the engineered safety features function of the Low Pressure Service Water System which supplies coolant to the reactor building coolers shall be made to demonstrate operability of the coolers. (M12)

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate valves have completed their travel, and fans are running at half speed. (See 3.6)

3 of 3

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

[1] A one-time extension of the Reactor Building Cooling system test frequency to a maximum of 23 months is allowed for Oconee Unit 2 during operating cycle 16. (See 3.6)

Add SR 3.7.8.1
SR 3.7.8.2
SR 3.7.8.3
SR 3.7.8.4

(A1 <except as marked>)

M28

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency	
1.	Control Rod Movement [1]	Movement of Each Rod	Monthly	<SEE 3.1>
2.	Pressurizer Safety Valves	Setpoint	18 months[4]	<SEE 3.4>
3.	Main Steam Safety Valves	Setpoint	18 months[4]	<SEE 3.7.1>
4.	Refueling System Interlocks[5]	Functional	Prior to Refueling	<SEE 3.3>
5.	Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly	<SEE 3.7.2>
6.	Reactor Coolant System [2] Leakage	Evaluate	Daily	<SEE 3.4>
7.	Emergency Condenser [6] Circulating Water System Test	Functional	18 months	
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	<SEE 3.7.1>
9.	Spent Fuel Cooling System	Functional	Prior to Refueling	
10.	High Pressure and Low [3] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	<SEE 3.5>
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months	<SEE 3.7.5>
12.	MSLB Feedwater Isolation [7] Feature	Functional	18 months	<SEE 3.7.3>
13.	Essential Siphon Vacuum [8] System Test	Functional	Quarterly (92 days)	

Supp 2

Supp 2

Supp 2

SR 3.7.8.9

M29

M29

SR 3.7.8.5
SR 3.7.8.6
SR 3.7.8.7
SR 3.7.8.8

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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Specification 3.7.8

(A1) <except as marked>

(1) Applicable only when the reactor is critical. – <SEE 3.1 + 3.7.2>

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure. – <SEE 3.4>

(3) Operating pumps excluded. – <SEE 3.5>

Supp 2 (4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. <SEE 3.4 + 3.7.1>

(5) Applicable only to the interlocks associated with the Reactor Building Purge System. – <SEE 3.3>

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System. LA10

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature. <SEE 3.7.3>

LCO Note (8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit.

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

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4 of 4

Specification 3.7.9

A1 Except as Marked

4.12 CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM

Applicability

Applies to control room pressurization and filtering system components

Objective

To verify that these systems and components will be able to perform their design functions.

Specification

4.12.1 Operating Tests

LA8

Suppl SR 3.7.9.1 a. Control room outside air booster fan system tests shall be performed quarterly. These tests shall consist of an external visual inspection, a flow measurement for each unit and pressure drop measurements across each filter bank. Pressure drop across pre-filter shall not exceed 1 inch H_2O and pressure drop across HEPA shall not exceed 2 inches H_2O. Fan motors shall be operated continuously for at least one hour, and all louvers shall be proven operable. (See 5.0)

LA11

Suppl 2 SR 3.7.9.1 b. Every 18 months, verify the system maintains the control room at a positive pressure with both outside air booster fans on during system operation.

4.12.2 Filter Tests

A12

Suppl SR 3.7.9.2 Every 18 months, for the Unit 1 and 2 and the Unit 3 control room an in-place leakage test using DOP on HEPA units and Freon-112 (or equivalent) on carbon units shall be performed at design flow on each filter train. Removal of 99.5 percent DOP by each entire HEPA filter unit and removal of 99.0 percent Freon-112 (or equivalent) by each entire carbon adsorber unit shall constitute acceptance performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing. (See 5.0)

Bases

The purpose of the control room pressurization filtering system is to protect the control room operators from the effects of accidental release of radioactive effluents or toxic gases in the Turbine Building or Auxiliary Building only. The system is designed with two 50 percent capacity filter trains each of which consists of a prefilter, high efficiency particulate filters, carbon filters, booster fans, air handling unit fans, and associated ductwork to pressurize the control room with outside air.

Since these systems are not normally operated, a periodic test is required to insure their operability when needed. Quarterly testing of this system will show that the system is available.

Suppl 2

Testing of the installed carbon adsorber stage and absolute filters every 18 months will verify the leak integrity of the cleanup system. Testing every 18 months will also verify the ability of the system to maintain the control room at a positive pressure to minimize infiltration of hazardous effluents.

A2

3 of 5

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Specification 3.7.9

4.14 REACTOR BUILDING PURGE FILTERS AND SPENT FUEL POOL VENTILATION SYSTEM

Applicability

Applies to testing of the Reactor Building purge filters for Units 2 and 3 and the spent fuel pool ventilation systems.

Objective

To verify that the Unit 2 and Unit 3 Reactor Building purge filters will per-form their design function and that when used with the spent fuel pool ventilation system, will reduce the off-site dose due to a fuel handling accident.

Specification

4.14.1 Operational and Performance Testing

Suppl. 2

a. Monthly, each train of the spent fuel pool ventilation system shall be operated through the respective Reactor Building purge filters for at least 15 minutes at design flow ± 10%.

b. Every 18 months, the spent fuel pool ventilation fans shall be shown to operate at design flow ± 10% when tested in accordance with ANSI N510-1975.

Suppl. 2

c. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the Reactor Building purge filters:

1. Every 18 months;

2. After each complete or partial replacement of HEPA filter bank or charcoal adsorber bank;

3. After any structural maintenance on the system housing;

4. After painting, fire, or chemical release in any ventilation zone communicating with the system.

d. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

(R4)

4 of 5

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Specification 3.7.9

Sect. 2 1

(R4)

c. Every 18 months, or following 720 hours of system operation, or after painting, fire, or chemical release in any ventilation zone communicating with the system, a carbon sample shall be removed from the Reactor Building purge filters for laboratory analysis. Within 31 days of removal, this sample shall be verified to show ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable.

Bases

The Unit 2 Reactor Building purge filter is used in the ventilation system for the common spent fuel pool for Units 1 and 2. The Unit 3 Reactor Building purge filter is used in the Unit 3 spent fuel pool ventilation system. Each filter is constructed with a prefilter, an absolute filter and a charcoal filter in series. The high efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine.

Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the doses for a fuel handling accident would be minimized.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the spent fuel pool ventilation system every month will demonstrate operability of the fans, filters and adsorber system.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

(A2)

5 of 5

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Specification 3.7.10

(AT Except as Marked)

4.5.4 Penetration Room Ventilation System

Applicability

Applies to testing of the Penetration Room Ventilation System

Objective

To verify that the Penetration Room Ventilation System is operable.

Specification

4.5.4.1 Operational and Performance Testing

(L8)

SR3.7.10.1 Surv.2

a. ~~Monthly~~ 31 days each train of the Penetration Room Ventilation System shall be operated for at least 15 minutes ~~at design flow ±10%~~.

Surv.1

b. Every 18 months, it shall be demonstrated that:

1. The Penetration Room Ventilation System fans operate at design flow (± 10%) when tested in accordance with ANSI N510-1975. (See 5.0)

SR3.7.10.2

2. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than six inches of water at the system design flow rate (± 10%). (A14)

SR3.7.10.3

3. Each ~~branch~~ Train of the Penetration Room Ventilation System is capable of automatic initiation.

SR3.7.10.5

4. The bypass valve for filter cooling ~~is manually operable~~ can be opened.

Surv.2

SR3.7.10.2

c. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the Penetration Room purge filters:

1. Every 18 months;

2. After each complete or partial replacement of a HEPA filter bank or charcoal adsorber bank;

3. After any structural maintenance on the system housing;

4. After painting, fire, or chemical release in any ventilation zone communicating with the system.

(See 5.0)

d. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

(A14)

2 of 3

~~Amendment No 228~~ (Unit 1)
~~Amendment No 229~~ (Unit 2)
~~Amendment No 225~~ (Unit 3)

(A1) Except as Marked

Specification 3.7.10

Add SR 3.7.10.2 — (A14)

Suppl. 2

SR 3.7.10.2

(See 5.0)

c. Every 18 months, or following 720 hours of system operation, or after painting, fire, or chemical release in any ventilation zone communicating with the system, a carbon sample shall be removed from the Penetration Room Ventilation system filters for laboratory analysis. Within 31 days of removal, this sample shall be verified to show ≥90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable.

(A15)

Suppl. 2

Bases

Pressure drop across the combined high efficiency particulate air (HEPA) filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per year operating cycle establishes performance capability.

(HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system every month will demonstrate operability of the filters and adsorber system. Operation for 15 minutes demonstrates operability and minimizes the moisture build up during testing.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

(A2)

Suppl. 2

Add SR 3.7.10.4 — (M14)

3 of 3

Section 3.7

LA/2

4.18 SNUBBERS

Applicability

Applies to hydraulic and mechanical snubbers used to protect the Reactor Coolant System and other safety-related systems.

Objective

To verify that the required hydraulic and mechanical snubbers are operable.

Specification

4.18.1 Each snubber associated with the Reactor Coolant System and other safety-related systems as specified in the appropriate Station Procedure shall be visually inspected. Visual inspections shall verify:

(1) that there are no visible indications of damage or impaired OPERABILITY,

(2) attachments to the foundation or supporting structure are secure, and

(3) in those locations where mechanical snubber movement can be manually induced, the snubbers shall be inspected as follows:

(a) Every 18 months, the inaccessible snubbers shall be inspected near the beginning and the end of an outage.

(b) In the event of a severe dynamic event, snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying freedom of motion using one of the following: (i) Manually induced snubber movement, (ii) evaluation of in place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced (or overhauled) before returning to power. Re-inspection shall subsequently be performed according to the schedule listed below.

Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.18.4. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be tested by starting with the piston at the as found setting

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Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

Section 3.7

LA12

and extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. Snubber operability will be verified in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	4 months ± 25%
5,6,7	2 months ± 25%
≥8	1 month ± 25%

Note: (1) The required inspection interval shall not be lengthened more than two steps per inspection.

(2) Snubbers may be categorized in two groups, "accessible" or "inaccessible," based on their accessibility during reactor operation. These two groups may be inspected independently according to be above schedule.

(3) Hydraulic and mechanical snubber inspection schedules are independent.

4.18.2 The seal service life of hydraulic snubbers shall be monitored to ensure that the seals do not exceed their expected service life by more than 10% between surveillance inspections. The maximum expected service life for the various seals, seal materials, and applications shall be estimated based on engineering information, and the seals shall be replaced so that the maximum expected service life is not exceeded by more than 10% during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.5.1.m.

4.18.3 At least once every 18 months, a representative sample, a minimum of 10% of the total of hydraulic snubbers in use in the plant, shall be functionally tested either in place or in a bench test. For each hydraulic snubber that does not meet the functional test acceptance criteria of Specification 4.18.4, an additional minimum of 10% of the hydraulic snubbers shall be functionally tested until none are found inoperative or all have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of hydraulic snubbers. The representative sample shall be selected randomly from the total population of safety-related hydraulic snubbers.

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4.18.6 Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions. Snubbers so exempted shall be listed in a permanent record which references the exemption letter date.

Bases

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval unless so determined, by the engineer, from a previous window of a schedule. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber.

Super 1

To provide assurance of snubber functional reliability, a representative sample of the installed hydraulic snubbers will be functionally tested every 18 months. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

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A20 A CTS provision comparable to the ITS Actions Note does not exist. This Action Note permits separate Condition Entry for each valve. CTS does not prohibit separate condition entry for each valve. Therefore this is an administrative change and is consistent with the NUREG.

A21 Not used.

2 A22 A CTS provision comparable to the Note to ITS 3.7.13 RA A.1 does not exist. This Note states that the provisions of LCO 3.0.3 does not apply. Although no explicit comparable CTS provision exists, it is unreasonable to apply CTS 3.0 since a shutdown of the applicable unit(s) provides no cure for the associated condition involving the boron concentration in the spent fuel pool. Additionally, no realistic scenario is envisioned which would preclude promptly completing the CTS specified actions, and therefore potentially requiring entry into CTS 3.0. Consequently, this change is considered administrative and is consistent with the NUREG.

A23 CTS 3.4.5 requires a specified volume of water be available in the UST, CST and HW. ITS 3.7.6 requires the UST, CST and HW be OPERABLE with OPERABILITY specified as containing the specified volume of water. ITS SR 3.7.6.1 specifies the required quantity of water consistent with the CTS 3.4.5 volume requirements. Therefore this change is an administrative change and is consistent with the NUREG.

A24 CTS Table 4.1-2, item number 3 requires testing main steam safety valves
2 every 18 months. The frequency of 18 months is modified by Note (4)
2 which states that the number of safety valves to be tested every 18
2 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years. ITS SR 3.7.1.1 requires verification of each required lift setpoints in accordance with the Inservice Testing Program at a frequency which is in accordance with the Inservice Testing Program. The Inservice Test Program requires setpoint testing in accordance with ASME Section XI testing requirements. Therefore this is an administrative change.

A25 CTS 4.8 requires testing the main steam stop valves. ITS SR 3.7.2.1 and SR 3.7.2.2 requires testing the turbine stop valves. The change in name is a presentation preference only since the components referred to are the same. This is an administrative change associated with a plant preference regarding equipment terminology.

A26 Not used

A27 CTS 3.5.7.2 requires the Main Feedwater control valve to be OPERABLE. ITS LCO 3.7.3 requires two MFCVs and two SFCVs to be OPERABLE. The paragraph under the applicability section in CTS 3.5.7 defines the main feedwater controls valves as both the Main Feedwater System main control valves and Main Feedwater startup control valves. The ONS design provides two Main Feedwater System main control valves and two Main

ITS Section 3.7

ID 111

Subject: Corrected Section 3.7, DOC A22 to refer to ITS 3.7.13 Required Action A.1.

A20 A CTS provision comparable to the ITS Actions Note does not exist. This Action Note permits separate Condition Entry for each valve. CTS does not prohibit separate condition entry for each valve. Therefore this is an administrative change and is consistent with the NUREG.

A21 Not used.

2 A22 A CTS provision comparable to the Note to ITS 3.7.13 RA A.1 does not exist. This Note states that the provisions of LCO 3.0.3 does not apply. Although no explicit comparable CTS provision exists, it is unreasonable to apply CTS 3.0 since a shutdown of the applicable unit(s) provides no cure for the associated condition involving the boron concentration in the spent fuel pool. Additionally, no realistic scenario is envisioned which would preclude promptly completing the CTS specified actions, and therefore potentially requiring entry into CTS 3.0. Consequently, this change is considered administrative and is consistent with the NUREG.

A23 CTS 3.4.5 requires a specified volume of water be available in the UST, CST and HW. ITS 3.7.6 requires the UST, CST and HW be OPERABLE with OPERABILITY specified as containing the specified volume of water. ITS SR 3.7.6.1 specifies the required quantity of water consistent with the CTS 3.4.5 volume requirements. Therefore this change is an administrative change and is consistent with the NUREG.

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A26 Not used

A27 CTS 3.5.7.2 requires the Main Feedwater control valve to be OPERABLE. ITS LCO 3.7.3 requires two MFCVs and two SFCVs to be OPERABLE. The paragraph under the applicability section in CTS 3.5.7 defines the main feedwater controls valves as both the Main Feedwater System main control valves and Main Feedwater startup control valves. The ONS design provides two Main Feedwater System main control valves and two Main

Feedwater startup control valves. This is an administrative requirement and is consistent with the NUREG.

A28 CTS 3.5.7.2.1.a permits a Main Feedwater control valve in one or more flow paths to be inoperable provided it is closed within 8 hours and verified to be once per 7 days. ITS Action A permits a MFCV to be inoperable provided it is closed or isolated within 8 hours and verified to be closed once per 7 days. ITS Action B permits a SFCV to be inoperable provided it is closed or isolated within 8 hours and verified to be once per 7 days. Since the MFCV and SFCV for each steam generator are in a different flow path, the CTS permits both a MFCV and a SFCV to be simultaneously inoperable for each steam generator. Therefore, this is an administrative change and is consistent with the NUREG.

A29 CTS 3.4.7 requires the main steam atmospheric dump valve flow path on each steam generator to be OPERABLE. ITS LCO 3.7.4 requires one ADV line per steam generator to be OPERABLE. The change from ADV flow path to ADV line is a presentation preference only. Therefore, this is an administrative requirement and is consistent with the NUREG.

A30 CTS 4.9.1 requires testing the EFW pumps in accordance with the requirements of CTS 4.0.4. CTS 4.0.4 states that Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components. ITS SR 3.7.5.2 requires verifying the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head. The SR 3.7.5.2 requirement to verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head is encompassed within the testing requirement required by ASME Section XI, subsection IWP. This change is an administrative change and is consistent with the NUREG.

A31 CTS 3.3.7.b states that tests or maintenance shall be allowed on any component of the LPSW system provided the redundant train of the LPSW system is operable. The entry condition for ITS Required Action A.1 is a required LPSW pump inoperable. With a required LPSW pump inoperable the redundant pump(s) and required flow path ensures the required safety function can be accomplished (albeit without an additional single failure). Although multiple flow paths exist from the discharge of the LPSW pumps to each required supplied component, some portions of the multiple flow paths are not totally redundant or independent. Consequently, the LPSW pumps and flow paths are not separated into trains.

A32 CTS Table 4.1-3, Note "*" clarifies that the SR Frequency is not applicable if reactor is in a cold shutdown condition for a period exceeding the sampling frequency. The provision of this Note regarding being in the cold shutdown condition for a period greater than the

sampling frequency is deleted since it is unnecessary. Since requirements of Specification 3.7.14 are not Applicable in MODE 5 (equivalent to CTS Cold Shutdown condition) the LCO and associated SRs are not applicable.

A33 CTS 3.8.13.a, 3.8.13.b, 3.8.13.c and 3.8.13.d specify requirements associated with moving casks in the SFP. The CTS presentation for these requirements states "Prior to spent fuel cask movement . . . ." or "Prior to dry storage transfer cask movement . . . ." ITS 3.7.15 is Applicable during movement of either the spent fuel shipping cask or the dry storage transfer cask in the SFP area. Although the CTS presentation can be interpreted to apply at any time prior to cask movement in the SFP, the extreme extension of such an interpretation makes the requirements applicable at all times. This is clearly not the intent since the purpose of these requirements is to ensure the spent fuel has decayed sufficiently at the time of cask movement in the SFP. This is considered a change in presentation only. No change in interpretation or application is intended. Therefore, this change is an administrative change.

A34 An explicit CTS requirement comparable to ITS 3.7.15 Action A does not exist. With requirements of the LCO not met, ITS 3.7.15 Action A requires immediately suspending movement of cask in the SFP area. Since CTS does not permit cask movement under this condition, CTS requires, although not explicitly, the same action. Therefore this change is considered administrative.

A35 An explicit CTS requirement comparable to ITS SR 3.7.15.1 does not exist. ITS SR 3.7.15.1 requires verification by administrative means that the decay time for fuel assemblies is within limit. The Frequency for this SR is "Prior to cask movement in the spent fuel." Although an explicit requirement comparable to ITS SR 3.7.15.1 does not exist, the very existence of CTS 3.7.15.1 imposes a requirement for at least an administrative verification of the decay times in the specified rows at some time prior to movement of the cask in the spent fuel area. This change is therefore considered administrative.

2 A36 CTS 4.5.2.1.2, Note 1 permits a one time extension of the test frequency
2 to a maximum of 23 months for Oconee Unit 2 during operating cycle 16.
2 This provision is no longer needed and is deleted since operating cycle
2 16 has passed. As such, the change is adminstrative.

ITS Section 3.7

ID 128

Subject: Revised Section 3.7 DOC A14 to also refer to CTS 4.5.4.1.b.1.

are not separately retained. As a result, this change is a presentation preference and is administrative in nature.

A12 CTS 4.12.2 provides details regarding testing the testing the CRVS filters. These details are relocated to the Ventilation Filter Testing Program (VFTP) specified in ITS 5.5.12. ITS SR 3.7.9.2 requires performance of the testing in accordance with the VFTP. Consequently, this is an administrative change and is consistent with the NUREG.

A13 Not used.

2 A14 CTS 4.5.4.1.b.1, 4.5.4.1.b.2, 4.5.4.1.c, 4.5.4.1.d and 4.5.4.1.e provides details regarding testing the PRVS filters. These details are relocated to the Ventilation Filter Testing Program (VFTP) specified in ITS 5.5.12 (see DOC LA 21). ITS SR 3.7.10.2 is added to require performance of the testing in accordance with the VFTP. Consequently, this is an administrative change and is consistent with the NUREG.

A15 CTS 4.5.4.1.e requires the PRVS system be declared inoperable if the laboratory analysis of the carbon does not meet acceptance criteria. This requirement is not separately retained in the ITS. ITS SR 3.0.1 states that failure to meet SR requirements constitutes failure to meet the LCO. If ITS SR 3.7.10.2 is not met, LCO 3.7.10 is not met. Failure to meet the LCO is equivalent to declaring the system inoperable. Consequently, this is an administrative change and is consistent with the NUREG.

A16 A CTS provision comparable to the Note to ITS 3.7.12 RA A.1 does not exist. This Note states that the provisions of LCO 3.0.3 does not apply. Although no explicit comparable CTS provision exists, it is unreasonable to apply CTS 3.0 since a shutdown of the applicable unit(s) provides no cure for the associated condition involving the boron concentration in the spent fuel pool. Additionally, no realistic scenario is envisioned which would preclude promptly completing the CTS specified actions, and therefore potentially requiring entry into CTS 3.0. Consequently, this change is considered administrative and is consistent with the NUREG.

A17 Not used.

A18 A CTS provision comparable to the Note to ITS SR 3.7.2.1 and the Note to SR 3.7.2.2 does not exist. These Notes state that the SRs are only required to be performed in MODES 1 and 2. The Notes permit the valves to be considered OPERABLE in MODE 3 without performance of SR 3.7.2.1 or SR 3.7.2.2. CTS does not require performance of surveillances prior to entering the mode of applicability. Therefore, the ITS provision to not require testing the valves when in MODE 3 is consistent with the CTS, administrative and is consistent with the NUREG.

A19 Not used.

ITS Section 3.7

ID 134

Subject: Revised SR 3.7.12.1 to retain CTS Frequency of 31 days and after each makeup to the Spent Fuel Pool. Corrected B 3.7.12 footer (AMENDMENT NOS. changed to Amendment Nos.)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify the spent fuel pool boron concentration is within limit.	31 days AND After each makeup to the spent fuel pool

2

B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

As described in the following LCO 3.7.13, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel pool racks in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ limits specified in the COLR, the criteria that limit the storage of a fuel assembly to specific rack locations are conservatively developed without taking credit for boron.

APPLICABLE SAFETY ANALYSES

The requirements for spent fuel pool boron concentration specified ensure that a minimum boron concentration is maintained in the pool. The requirements for spent fuel assembly storage specified ensure that the fuel stored in the pool remains subcritical. The water in the spent fuel storage pool normally contains soluble boron which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines (Ref. 2) based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of the spent fuel storage racks is based on the use of unborated water, which maintains the spent fuel pool in a subcritical condition during normal operation with the pool fully loaded. The double contingency principle discussed in ANSI N-16.1-1975, (Ref. 1) and the NRC guidelines (Ref. 2) allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the accidental misloading of a fuel assembly. This could increase the reactivity of the spent fuel pool. To mitigate this postulated criticality related accident, boron is dissolved in the pool water.

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 and 3 of 10 CFR 50.36 (Ref. 3).

(continued)

BASES (continued)

LCO

The concentration of dissolved boron in the spent fuel pool specified in the COLR preserves the assumption used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS

A.1, A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position. Immediate action is also required to initiate action to restore the SFP boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 31 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1 (continued)

short period of time and verification is required after each
2 makeup to the SFP. The verification after each makeup
2 should be completed within 12 hours after a 24 hour
2 recirculation period to allow for mixing. This Completion
2 Time is appropriate since no major replenishment of pool water is expected to take place over this period.

REFERENCES

1. ANSI N-16.1-1975.

2. Letter from B.K. Grimes (USNRC) to Power Reactor Licensees dated April 14, 1978.

3. 10 CFR 50.36.

(A1) (except as marked)

Table 4.1-3

Minimum Sampling Frequency And Analysis Program

	Item	Check	Frequency
1.	Reactor Coolant	a. Gamma Isotopic Analysis	a. 3 times/week* (SEE 3.4)
		b. Boron Concentration	b. 2 times/week** (SEE 3.9)
		c. $\bar{E}$ Determination (2)	c. Semi-annually
2.	Borated Water Storage Tank Water Sample	Boron Concentration	Weekly* and after each makeup (SEE 3.5)
3.	Core Flooding Tank	Boron Concentration	Monthly* and after each makeup
4.	Spent Fuel Pool Water Sample	Boron Concentration	31 days Monthly and after each makeup (Supp 2, SR 3.7.12.1)
5.	OTSG or Final Feedwater	Gamma Isotopic Analysis	Weekly* (SEE 3.7.14)
6.	Concentrated Boric Acid Tank	Boron Concentration	Weekly* (see 3.5)

* Not applicable if reactor is in a cold shutdown condition for a period exceeding the sampling frequency. (SEE 3.4, 3.5 + 3.7.14)

** Applicable only when fuel is in the reactor. (SEE 3.9)

3 of 3

Amendment No. 221 (Unit 1)
Amendment No. 221 (Unit 2)
Amendment No. 218 (Unit 3)

Specification 3.7.12

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Verify the spent fuel pool boron concentration is within limit.	~~7 days~~

T4.1-3 Item 4

31 days

AND

After each makeup to the spent fuel pool

32

Supp 2

BASES

TSTF-70, R1

The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. Alternatively, beginning a verification of the spent fuel pool locations to ensure proper location of the fuel, can be performed. However, prior to movement of fuel assemblies, the concentration of boron will be restored

APPLICABILITY (continued)

verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movement in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

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

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is ~~most efficiently~~ achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position.

Insert B3.7-74A

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

SR 3.7.12.1

SURVEILLANCE REQUIREMENTS

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The ~~7~~ 31 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

Insert B3.7-74B

and verification is required after each makeup to the SFP.

REFERENCES

~~None.~~

1. ANSI N-16.1-1975
2. Letter from B.K. Grimes (USNRC) to Power Reactor Licensees dated April 14, 1978.
3. 10 CFR 50.36

Insert B3.7-74A

Immediate action is also necessary to initiate action to restore the SFP boron concentration to within limits.

Insert B3.7-74B

2 The verification after each makeup should be completed within 12 hours after a
2 24 hour recirculation period to allow for mixing. This Completion Time is appropriate since no major replenishment of pool water is expected to take place over this period.

29 NUREG 3.7.11 is not adopted. The ONS current licensing basis does not credit the Chilled Water System (CWS) in the accident analyses. Additionally, the CTS does not contain any requirements for the CWS. Therefore, controls for this system are administrative and not adopted in the ITS.

30 NUREG Specification 3.7.13, Fuel Storage Pool Ventilation System (FSPVS) is not adopted. Although the ONS design provides a comparable system, the Spent Fuel Pool Ventilation System (SFPVS), and the CTS includes comparable requirements, the NUREG specification is not adopted because the SFPVS has been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provide in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of the Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Stations Unit 1, 2, and 3 Technical Specifications."

31 Not used.

32 The Applicability for NUREG 3.7.15 and the Frequency for SR 3.7.15.1 are modified to reflect the current licensing basis. Specifically, the NUREG provision regarding the modification of Applicability when a spent fuel pool (SFP) verification has been completed after the last movement of fuel assemblies in the SFP is not adopted. Related NUREG Required Action A.2.2 is also not adopted. The ONS current licensing basis includes consideration of other events such as dropping the shipping cask into the SFP which are not excluded by the SFP verification.

The current licensing basis regarding frequency of verifying the spent fuel pool boron concentration is retained. The periodic frequency of 31 days is considered adequate since the event specific frequency of "after each makeup to the spent fuel pool . . . " provides reasonable assurance the required boron concentration is maintained for the only expected event that can result in decreasing the boron concentration in the spent fuel pool.

33 NUREG Specification 3.7.16 is modified to reflect the current licensing basis regarding the requirements applicable to spent fuel storage locations in both the spent fuel pool shared between the Unit 1 and Unit 2 and the Unit 3 spent fuel pool. This change is necessary to encompass the CTS requirements for the three ONS units and the two spent fuel pools.

34 The current licensing basis for main feedwater isolation valves does not include the main feedwater stop valves (MFSV). The MFSVs are not assumed to close in the accident analysis. Closure of both the main feedwater control valves (MFCV) and both startup feedwater control valves (SFCV) is assumed in the accident analysis. Therefore, NUREG Specification 3.7.3 is modified to remove ACTIONS associated with the MFSVs (ACTION A. ACTION D is eliminated since the ONS current licensing basis does not require two valves in the same flow path to be OPERABLE.

35 Not used.

36 Not used.

37 The Note to NUREG 3.7.4 Action A is not adopted. This Note permits entry into a MODE or other specified condition of the LCO's Applicability when the LCO is not met. Since the ADVs are credited in certain small break LOCA analysis, it is considered inappropriate to permit changes in MODE or other specified conditions of the LCO's Applicability when the LCO is not met.

38 Not used.

39 TSTF-100, which modifies NUREG 3.7.4 Required Action B.1 to state, "Restore all but one ADV line to OPERABLE status." is not adopted. The change represented by TSTF-100 is potentially confusing and unnecessary, since ONS has only two ADV lines. The TSTF-100 modified Required Action implies more than two ADV lines are provided.

40 The Note to SR 3.7.5.2 is not adopted. The ONS design permits testing the turbine driven EFW pump using auxiliary steam supplied from another ONS unit or the plant auxiliary boiler.

41 Not used.

42 The Applicability to NUREG 3.7.14 is expanded to include "During movement of a cask over the spent fuel pool." Action A is modified to also require suspending movement of the cask over the spent fuel pool. The Note to 3.7.14 Required Action A.1 is moved to precede both Required Action A.1 and Required Action A.2. This is necessary to reflect the current licensing basis for Oconee. ONS is licensed to permit placing a shipping cask in the spent fuel pool. The water level in the spent fuel pool is an initial assumption for the cask drop accident.

43 Not used.

44 Not used.

45 NUREG SR 3.7.2.1 is modified to reflect the TSV closure time associated with closure channel A. A second surveillance, SR 3.7.2.2, is added to verify the TSV closure time associated with channel B. The ONS design

provides two means of closing each TSV. Each closure mechanism results in different closure times. Closure channel A actuates the "disk dump" closure mechanism which closes the associated TSV in ≤ 1 second. Closure channel B actuates the closure mechanism utilizing a restricting orifice and closes the associated TSV in ≤ 15 seconds. These closure times are consistent with the current licensing basis for TSV closure.

46 The ONS design uses a portion of the Condenser Circulating Water System to provide a flow path from Lake Keowee to the suction of the Low Pressure Service Water Systems by utilizing "siphon headers". Siphon header OPERABILITY requires associated vacuum pumps and seal water supply headers. NUREG Specification 3.7.9 is modified to encompass the CLB requirements for the ECCW siphon headers.

47 The current licensing basis permits 12 hours to place a unit in Hot Shutdown (equivalent to ITS MODE 3) and 60 hours to place the Unit in a condition outside the Applicability of the Specification when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 and 60 hours to place the Unit in a condition outside the Applicability of the specification is retained in the ITS.

48 Consistent with the CLB, Lake Keowee water level limits are specified in UFSAR Chapter 16. NUREG SR 3.7.9.1 is modified to require verifying lake level within limits. The ITS Bases for this SR refers to UFSAR Chapter 16 for the actual limits. Lake level requirements are maintained in UFSAR Chapter 16 since the values are subject to change resulting from modifications and changes in operating practices, which may impact LPSW System flow requirements.

49 ITS SR 3.7.8.1 is added to require periodic verification that required ESV pumps are in operation. Periodic verification that required ESV pumps are operating is appropriate since operation of required ESV pumps is a requirement for OPERABILITY of an ECCW siphon header. ITS SR 3.7.8.4 is adopted to provide for periodic verification that non-automatic valves in the ECCW siphon header, required ESV and required SSW flow paths are in the correct position. Periodic verification that valves in the ECCW siphon header, required ESV and required SSW flow paths are in the correct position is appropriate since correct valve positions are a requirement for OPERABILITY of an ECCW siphon header. Consistent with the CLB, ITS SR 3.7.8.5, SR 3.7.8.6, SR 3.7.8.7, SR 3.7.8.8 and SR 3.7.8.9 are adopted to retain requirements encompassed in CTS Table 4.1-2, Items 7 and 13.

50 Bracketed NUREG SR 3.7.9.3 is not adopted. NUREG SR 3.7.9.3 requires operation of each cooling tower fan every 31 days. The ONS design does not include cooling towers. Consequently, there are no cooling tower fans.

51 The brackets are removed and the Completion Time for NUREG 3.7.2 Required Actions A.1 and C.1 of 8 hours is adopted. There is no CLB for these value since a similar CTS requirement does not exist. Eight hours is considered reasonable considering the low probability of an event in this time period and considering the TSVs isolate a closed system which provides an additional barrier against releases.

52 The brackets are removed and the plant specific value of one provided in LCO 3.7.4. The ONS design provides one atmospheric dump valve line per steam generator.

53 Bracketed NUREG SR 3.7.10.5 is not adopted since there is no comparable CTS requirement. (NUREG-1430 provides no Bases information for this SR.)

54 Not used.

55 The Applicability for NUREG 3.7.16 is modified to eliminate the bracketed reference to Region 2. Although restrictions upon storage location exist in the spent fuel pool, the LCO is applicable to storage anywhere in the spent fuel pool.

58 The portion of NUREG SR 3.7.1.1 requiring MSRVs settings to be within 1% of the setpoint following testing is not adopted. This modification to the SR is made to reduce the potential for confusion. The current licensing basis for ONS requires the MSRVs setting to be within 1% of the setpoint at all times. Retention of the deleted wording could imply that the settings may be within some different tolerance band at other times.

59 The brackets are removed and the closure time specified in NUREG SR 3.7.3.1 is changed to 25 seconds. The value of 25 seconds for closure of the MFCVs and SFCVs is consistent with current plant analysis.

60 The brackets are removed and the Frequency for NUREG 3.7.4.2 is adopted. There is no CTS value for SR 3.7.4.2 since it is not a current requirement. The Frequency of 18 months for the surveillance of the ADV block valve is reasonable and is consistent with the Frequency for the ADV valves.

61 The second entry Condition for NUREG 3.7.9 Action B is not adopted. The current licensing basis for one inoperable ECCW siphon allows 60 hours to be in cold shutdown. This Completion Time is retained for ITS 3.7.8 Required Action B.2. The Completion Time of 60 hours is not considered to be appropriate for the second entry Condition in NUREG 3.7.9 Action B. There is no CTS provision for this condition. With the ECCW inoperable for reasons other than an inoperable ECCW siphon, CTS 3.0 applies requiring the unit be placed in Hot Shutdown within 12 hours and cold shutdown in 36 hours. For this Condition, the non-adoption of the

2 M14 A CTS requirement comparable to ITS SR 3.7.10.4 does not exist. SR 3.7.10.4 requires verification at a Frequency of 18 months on a staggered test basis that each Penetration Room Ventilation System (PRVS) train can maintain a specified negative pressure relative to atmospheric pressure when operated at a specifed flow rate. This SR verifies the integrity of the penetration room areas. The ability of the PRVS to maintain a negative pressure, with respect to outside atmosphere, is periodically tested to verify proper functioning of the PRVS. During the post accident mode of operation, the PRVS is designed to maintain a slight negative pressure in the penetration rooms with respect to outside atmosphere to prevent unfiltered LEAKAGE. The PRVS is designed to maintain this negative pressure at a flow rate of 1000 ± 10% cfm from the area. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs. Requiring a periodic verification the PRVS can maintain the required penetration room negative pressure is a more restrictive requirement upon unit operation and is consistent with the NUREG.

2 M15 Not used.

M16 A CTS surveillance requirement comparable to ITS SR 3.7.5.5 does not exist. This SR ensures that the EFW System is properly aligned by verifying the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of valves has occurred. This SR is appropriate and is an acceptable restriction upon operation since it ensures that the flow path from the CST to the steam generator is properly aligned. This change is consistent with the NUREG.

M17 Not used.

M18 If the control valves (MFCVs and SFCVs) are not restored to OPERABLE status within 72 hours CTS 3.5.7.2.1.b requires the unit be placed in Hot Shutdown within 12 hours and with RCS temperature less than 250°F within an additional 18 hours (total of 30 hours). ITS 3.7.3 RA C.1 requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. Requiring the unit be placed in MODE 4 within 18 hours is an additional restrictions upon unit operation. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

M19 CTS 3.4.1 requires the CST, UST and HW to meet specified requirements when RCS temperature is > 250°F. ITS 3.7.6 APPLICABILITY requires the CST, UST and HW to meet specified requirements when in MODES 1, 2, & 3 and in MODE 4 when a steam generator is relied upon for heat removal. Requiring the CST, UST and HW to be OPERABLE in MODE 4 when SGs are

ITS Section 3.7

ID 137

Subject: Minor editorial changes - defined ESV, CCW & SSW in SR 3.7.8.1, 3, & 6, corrected SR numbering on page B 3.7-9, corrected header on B 3.7-27 & 28 to state "CST, UST and HW", added double line to top of 3.7.09 ACTIONS Table & added comma to 3.7.11 Applic.

3.7 PLANT SYSTEMS

3.7.8 Emergency Condenser Circulating Water (ECCW) System

LCO 3.7.8 Two ECCW siphon headers shall be OPERABLE.

----------NOTE----------
Not applicable on each Unit until after completion of the Service Water upgrade modifications on the respective Unit.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ECCW siphon header inoperable.	A.1	Restore required ECCW siphon header to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 5.	60 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
2 2	SR 3.7.8.1	Verify required Essential Siphon Vacuum (ESV) pumps are in operation.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.8.2	Verify Keowee Lake water level is within limits.	24 hours
SR 3.7.8.3	Verify average water temperature of Condenser Circulating Water (CCW) inlet is $\leq$ 90°F.	24 hours
SR 3.7.8.4	Verify each manual and non-automatic power operated valve in each ECCW siphon header flow path, required ESV flow paths and required SSW flow paths that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.7.8.5	Verify upon an actual or simulated actuation signal each ESV float valve actuates to the correct position.	92 days
SR 3.7.8.6	Verify upon an actual or simulated actuation signal each required ESV and Siphon Seal Water (SSW) valve actuates to the correct position.	92 days
SR 3.7.8.7	Verify the developed suction head of each required ESV pump at the test point is greater than or equal to the required developed suction head.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.8.8	Verify each required ESV pump automatically starts in $\leq$ 1200 seconds upon an actual or simulated restoration of emergency power.	92 days
SR 3.7.8.9	----------NOTE---------- Not required to be performed for Units 1 and 2 with the shared Unit 1 and 2 LPSW System taking suction from the siphon. ---------- Verify upon an actual or simulated trip of the CCW pumps and ESV pumps that the rate of water level drop in the ECCW siphon header is within limits.	18 months

3.7 PLANT SYSTEMS

3.7.9 Control Room Ventilation System (CRVS) Booster Fans

LCO 3.7.9 Two CRVS Booster Fan trains shall be OPERABLE.

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

2 | ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Control Room pressure ≤ 0.0 psig during operation of two CRVS Booster Fan trains.	A.1	Restore Control Room pressure to > 0.0 psig during operation of two CRVS Booster Fan trains.	30 days
B. One CRVS Booster Fan train inoperable for reasons other than Condition A.	B.1	Restore CRVS Booster Fan train to OPERABLE status.	72 hours
C. Two CRVS Booster Fan trains inoperable for reasons other than Condition A.	C.1	Restore one CRVS Booster Fan train to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1 AND	Be in MODE 3.	12 hours
	D.2	Be in MODE 5	36 hours

3.7 PLANT SYSTEMS

3.7.11 Spent Fuel Pool Water Level

LCO 3.7.11 The Spent Fuel Pool water level shall be ≥ 21.34 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the Spent Fuel Pool, 2
During movement of cask over the Spent Fuel Pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent Fuel Pool water level not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	A.1 Suspend movement of irradiated fuel assemblies in Spent Fuel Pool.	Immediately
	AND	
	A.2 Suspend movement of cask over Spent Fuel Pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify the Spent Fuel Pool water level is ≥ 21.34 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is 18 months. The 18 month Frequency to demonstrate valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3, with the unit at operating temperature and pressure, as discussed in the Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

2

SR 3.7.2.2

This SR verifies that TSV closure time of each TSV is $\leq$ 15.0 seconds on an actual or simulated actuation signal from Channel B. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is 18 months. The 18 month Frequency to demonstrate valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE 3, with the unit at operating temperature and pressure, as discussed in the Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The CST, UST, and HW provides cooling water to remove decay heat following events in the accident analysis, as discussed in the UFSAR, Chapters 10 and 15 (Refs. 2 and 3, respectively).

The water inventory from the CST will not be available during a station blackout due to unavailability of power to the CST transfer pumps.

The required inventory in the UST, CST, and HW has not been specifically analyzed regarding the capability to permit cooling the unit down and transferring to the decay heat removal loops. The required inventory permits maintaining EFW capability until either cooling capability using the main condenser can be restored or until the Standby Shutdown Facility (SSF) (LCO 3.10.1, "Standby Shutdown Facility (SSF)") is placed in service.

The CST, UST, and HW satisfy Critera 2 and 3 of 10 CFR 50.36 (Ref. 4).

LCO

To satisfy LCO requirements, the UST, CST, and HW must contain the specified volume of water available to the EFW System.

The OPERABILITY of CST, UST, and HW is determined by maintaining the tank volume at or above the minimum required volume.

APPLICABILITY

In MODES 1, 2, 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST, UST, and HW are required to be OPERABLE.

In MODES 5 and 6, the CST, UST, and HW are not required because the EFW System is not required.

ACTIONS

A.1 and A.2

If the requirements of the LCO are not met, the unit must be placed in a MODE in which the LCO does not apply, with the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 4, without reliance on steam generators for heat removal, within 24 hours. This allows an additional 6 hours for the DHR System to be placed in service after entering MODE 4.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST, UST, and HW contain the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST, UST, and HW inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms to alert the operator to abnormal deviations in CST, UST, and HW levels.

REFERENCES

1. UFSAR, Section 10.4.
2. UFSAR, Chapter 10.
3. UFSAR, Chapter 15.
4. 10 CFR 50.36.

ITS Section 3.6 3.7 4.0 5.0

ID 139

Subject: Revised ITS to remove Penetration Room Ventilation System Tech Spec change 95-04

B 3.7 PLANT SYSTEMS

B 3.7.10 Penetration Room Ventilation System (PRVS)

BASES

BACKGROUND

The PRVS filters air from the area of the active penetration rooms during the recirculation phase of a loss of coolant accident (LOCA).

The PRVS consists of two independent, redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated carbon adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the Reactor Building penetration rooms area following receipt of an Engineered Safeguards actuation signal (ESAS).

The PRVS is a standby system. During emergency operations, the PRVS valves are realigned, and fans are started to begin filtration. Upon receipt of the ESAS signal(s), the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and carbon adsorbers.

The PRVS is discussed in the UFSAR, Sections 6.5.1, 9.4.7, and 15.4.7 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

2
2

The design basis of the PRVS is established by the Maximum Hypothetical Accident (MHA). In such a case, the system limits radioactive releases to within 10 CFR 100 (Ref. 7) requirements and personnel doses in the Control Room are maintained within the limits of 10 CFR 20 (Ref. 4). The analysis of the effects and consequences of an MHA is presented in Reference 3. No credit is taken in the analysis for any reduction in Control Room Dose provided by the PRVS filters.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.7.
3. UFSAR, Section 15.4.7.
4. 10 CFR 20.
5. 10 CFR 50.36.
6. Regulatory Guide 1.52.
2 7. 10 CFR 100.

Specification 3.7.10

(AT) Except as marked

Add SR 3.7.10.2 (A14)

(See 5.0)

Suppl. 2

SR 3.7.10.2

c. Every 18 months, or following 720 hours of system operation, or after painting, fire, or chemical release in any ventilation zone communicating with the system, a carbon sample shall be removed from the Penetration Room Ventilation system filters for laboratory analysis. Within 31 days of removal, this sample shall be verified to show ≥90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable.

(A15)

Suppl. 2

Bases

Pressure drop across the combined high efficiency particulate air (HEPA) filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per year operating cycle establishes performance capability.

(HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system every month will demonstrate operability of the filters and adsorber system. Operation for 15 minutes demonstrates operability and minimizes the moisture build up during testing.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

(A2)

Suppl. 2

Add SR 3.7.10.4 (M14)

3 of 3

2 M14 A CTS requirement comparable to ITS SR 3.7.10.4 does not exist. SR 3.7.10.4 requires verification at a Frequency of 18 months on a staggered test basis that each Penetration Room Ventilation System (PRVS) train can maintain a specified negative pressure relative to atmospheric pressure when operated at a specifed flow rate. This SR verifies the integrity of the penetration room areas. The ability of the PRVS to maintain a negative pressure, with respect to outside atmosphere, is periodically tested to verify proper functioning of the PRVS. During the post accident mode of operation, the PRVS is designed to maintain a slight negative pressure in the penetration rooms with respect to outside atmosphere to prevent unfiltered LEAKAGE. The PRVS is designed to maintain this negative pressure at a flow rate of 1000 ± 10% cfm from the area. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs. Requiring a periodic verification the PRVS can maintain the required penetration room negative pressure is a more restrictive requirement upon unit operation and is consistent with the NUREG.

2 M15 Not used.

M16 A CTS surveillance requirement comparable to ITS SR 3.7.5.5 does not exist. This SR ensures that the EFW System is properly aligned by verifying the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of valves has occurred. This SR is appropriate and is an acceptable restriction upon operation since it ensures that the flow path from the CST to the steam generator is properly aligned. This change is consistent with the NUREG.

M17 Not used.

M18 If the control valves (MFCVs and SFCVs) are not restored to OPERABLE status within 72 hours CTS 3.5.7.2.1.b requires the unit be placed in Hot Shutdown within 12 hours and with RCS temperature less than 250°F within an additional 18 hours (total of 30 hours). ITS 3.7.3 RA C.1 requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. Requiring the unit be placed in MODE 4 within 18 hours is an additional restrictions upon unit operation. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

M19 CTS 3.4.1 requires the CST, UST and HW to meet specified requirements when RCS temperature is > 250°F. ITS 3.7.6 APPLICABILITY requires the CST, UST and HW to meet specified requirements when in MODES 1, 2, & 3 and in MODE 4 when a steam generator is relied upon for heat removal. Requiring the CST, UST and HW to be OPERABLE in MODE 4 when SGs are

relied upon for decay heat removal is a more restrictive requirement upon unit operation and is consistent with the NUREG. Requiring the UST, CST and HW to be OPERABLE in MODE 4, when a steam generator is relied upon for heat removal, ensures a water supply is available to the EFW system when the EFW system is required to be OPERABLE.

M20 The CTS 3.13 limit for secondary system activity is 1.4 μCi/cc I-131, which is roughly equivalent to 2.0 μCi/gm DOSE EQUIVALENT I-131. This limit has been determined to be inappropriately high. ITS 3.7.14 specifies a limit of 0.10 μCi/gm Dose Equivalent I-131. A value of 0.10 μCi/gm Dose Equivalent I-131 is assumed in the reanalysis of UFSAR Chapter 15 accidents and transients. A topical report regarding the UFSAR Chapter 15 reanalysis methodology is being submitted separately for NRC approval and is described in the transmittal letter associated with the ONS ITS conversion. Therefore this is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M21 CTS 3.15.1 requires the PRVS to be OPERABLE whenever containment integrity is required. CTS 3.6.1 requires containment integrity whenever all three following conditions exist:

a.- reactor coolant pressure is $\geq$ 300 psig,
b.- reactor coolant temperature is $\geq$ 200°F, and
c.- nuclear fuel is in the core.

With these criteria, containment integrity is required sometime during ITS MODE 4 but not necessarily when this MODE was entered. The Applicabilities of ITS 3.7.10 for the PRVS include MODES 1, 2, 3, and 4. As such, the change represents an additional restriction on unit operation. This change is consistent with the NUREG. In MODES 1, 2, 3, and 4, the PRVS is required to be OPERABLE consistent with the OPERABILITY requirements of the containment.

M22 A CTS provision comparable to ITS 3.7.5 Action F does not exist since the CTS does not include operability requirements upon the EFW when the unit is less than 250°F. If the required EFW pump and required EFW flow path are not OPERABLE in MODE 4 when the steam generators are relied upon for decay heat removal, ITS 3.7.5 Action F requires action be initiated immediately to restore the required MDEFW pump and required EFW flow path to OPERABLE status. In this Condition, the Required Action is necessary to assure the steam generators can continue to be relied upon for decay heat removal. This change represents a more restrictive requirement upon unit operation.

M23 If one Atmospheric Dump Valve (ADV) is not restored to OPERABLE status within 7 days or if both ADVs are not restored to service within 24 hours, CTS 3.4.7.a and CTS 3.4.7.b require the unit be placed Hot Shutdown within 12 hours and below 350°F within another 24 hours. With Required Action and Associated Completion Time not met, ITS 3.7.4 Condition C requires the unit be placed in MODE 3 within 12 hours and in MODE 4 (> 200°F and < 250°F), without reliance upon a steam generator for heat removal, within 24 hours. The requirements to place the unit

in a condition without reliance upon a steam generator for heat removal and to be in this condition within 24 hours (instead of 36 hours) are more restrictive requirements upon unit operation and are consistent with the NUREG. Placing the unit in a condition without reliance upon a steam generator for heat removal is necessary to insure heat removal capability if the heat removal capability provided by the condenser become unavailable. The 24 hour Completion Time is reasonable based upon operating experience to reach the condition in an orderly manner without challenging plant systems.

M24 A CTS requirement comparable to ITS SR 3.7.4.2 does not exist. ITS SR 3.7.4.2 requires stroking the ADV block valves through one complete cycle at an 18 month frequency. Cycling the valves open and closed demonstrates their ability to perform their function. This additional requirement is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M25 CTS requires the Control Room booster fan trains to be OPERABLE above Hot Shutdown (i.e., when in ITS MODES 1 and 2). The Applicability of ITS LCO 3.7.9 is MODES 1, 2, 3 and 4. In MODES 1, 2, 3 and 4, the Control Room booster fan trains must be OPERABLE to reduce the radiation dose to personnel in the Control Room during and following an accident.

If the Control Room booster fan trains cannot be restored to OPERABLE status within the specified times, CTS 3.15.2.a and 3.15.2.b requires the unit be placed in Hot Shutdown within 12 hours. In this Condition ITS 3.7.9 RA D.2 requires the unit be placed in MODE 5 within 36 hours. Placing the unit in MODE 5 within 36 hours is necessary to place the unit in a condition outside the Applicability of ITS LCO 3.7.9. This change is a more restrictive requirement upon unit operation and is consistent with the NUREG.

M26 A specification comparable to ITS specification 3.7.11 does not exist. ITS specification 3.7.11 provides requirements for a minimum water level in the spent Fuel Pool. ITS LCO 3.7.11 requires the water level in the SFP to be 21.34 feet above the water level of the top of the irradiated fuel assemblies seated in the fuel storage racks. The minimum water level in the Spent Fuel Pool is consistent with the assumption of iodine decontamination factors following a fuel handling or cask drop accident. The Applicability for ITS Specification 3.7.11 is during movement of irradiated fuel assemblies in the spent fuel pool and during movement of the cask over the spent fuel pool. This LCO applies during movement of irradiated fuel assemblies in the Spent Fuel Pool or movement within the Spent Fuel Pool or movement of the cask over the Spent Fuel Pool since the potential for a release of fission products exists at these times. With the SFP water level not within limits, ITS 3.7.11 RA A.1 requires immediately suspending movement of irradiated fuel assemblies in the SFP and the cask over the SFP. When the initial conditions for an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the Spent Fuel Pool at less than the required level, the movement of fuel assemblies in the Spent Fuel Pool is

immediately suspended. This effectively precludes the occurrence of a fuel handling accident. ITS SR 3.7.11.1 requires periodic verification that the SFP water level is ≥ 21.34 feet above the top of irradiated fuel assemblies seated in the storage racks. This SR verifies that sufficient Spent Fuel Pool water is available in the event of a fuel handling or cask drop accident. The water level in the Spent Fuel Pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable.

The addition of this specification is a more restrictive requirement upon plant operation and is consistent with the NUREG. This specification ensures the water level in the Spent Fuel Pool is consistent with the assumptions of the safety analysis for fuel handling and cask drop accident.

M27 With less than the specified number of MSRVs OPERABLE, CTS Actions are specified by CTS 3.0. CTS 3.0 requires the unit to be in Hot Shutdown within 12 hours and cold shutdown within 36 hours. However, since the CTS applicability for the MSRVs is with RCS temperature > 250°F, CTS 3.0 only requires the RCS temperature be reduced to ≤ 250°F (equivalent to ITS MODE 4) at which point the CTS requirements for MSRVs is no longer applicable. Consequently, CTS 3.0 requires the unit be placed in MODE 4 within 36 hours. With less than the specified number of MSRVs OPERABLE, ITS 3.7.1 Action A requires the unit be placed in MODE 3 within 12 hours and MODE 4 within 18 hours. The requirement to be in MODE 4 within 18 hours instead of 36 hours is a more restrictive requirement upon unit operations and is consistent with the NUREG. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

M28 CTS requirements comparable to ITS SR 3.7.8.1, SR 3.7.8.2, SR 3.7.8.3 and SR 3.7.8.4 do not exist. SR 3.7.8.1 requires periodic verification that the required ESV pumps are in operation. This ensure that air from inleakage and degassing is removed to ensure the capability to establish siphon flow. SR 3.7.8.2 verifies periodic verification that Keowee Lake level is within limit, which ensures ECCW siphons can provide sufficient flow to ensure adequate NPSH is available for operating the LPSW pumps. SR 3.7.8.3 requires periodic verification that CCW inlet temperature is ≤ 90°F. This ensure inlet water to the LPSW system is consistent with assumptions. SR 3.7.8.4 provides periodic verification the correct alignment for manual, and non-automatic power operated valves in the ECCW siphon header flow paths, required ESV flow paths and required SSW flow paths. These SRs provides assurance that the proper conditions (i.e., flowpath, pump status, lake level and water temperature) exist for ECCW siphon header operation.

M29 CTS Table 4.1-2, Item 7 requires performance of am Emergency Condenser Circulating Water System Functional Test. ITS SR 3.7.8.9 verifies upon a simulated or actual loss of power to the CCW pumps and ESV pumps that the rate of water level decrease in the CCW piping is within limits.

This ensures air inleakage and degassing is within the removal capabilities of the ESV pumps. Limiting air inleakage is necessary to ensure the ECCW siphons headers can provide sufficient NPSH to the LPSW pumps.

CTS Table 4.1-2, Item 13 requires performance of an Essential Siphon Vacuum System Functional Test. ITS SR 3.7.8.5, SR 3.7.8.6, SR 3.7.8.7 and SR 3.7.8.8 provides requirements comparable to CTS Table 4.1-2, Item 13. SR 3.7.8.4 verifies that upon an actual or simulated actuation signal each ESV float valve actuates to the correct position. SR 3.7.8.5 verifies that upon an actual or simulated actuation signal each required ESV and SSW valve actuates to the correct position. SR 3.7.8.6 verifies that the developed suction head of each required ESV pump at the test point is ≥ the required developed suction head. SR 3.7.8.7 verify each required ESV pump starts within specified time period upon an actual or simulated restoration of emergency power. These SRs ensure the ESV and SSW Systems adequately support ECCW siphon header function. The ECCW siphon headers function to provide sufficient NPSH to the LPSW pumps.

The comparable ITS SR requirements are more prescriptive and therefore are more restrictive requirements upon unit operation.

M30 An explicit CTS requirement comparable to ITS SR 3.7.3.1 does not exist. ITS SR 3.7.3.1 requires verifying valve closure time is less than 25 seconds. A Note to SR 3.7.3.1 states the SR is only required to be performed in MODES 1 and 2. Although the stroke time for these valves is verified in accordance with the IST program, the inclusion of the stroke time in the ITS is a more restrictive requirement upon unit operation and is consistent with the NUREG. The inclusion of the SR Note is more restrictive since CTS does not require performance of SRs prior to entry into the applicability for the specification. The addition of the ITS SR Note is consistent with the NUREG. The inclusion of the stroke time is acceptable since it ensures equipment operation is consistent with the assumptions in the applicable safety analysis. The inclusion of the Note is acceptable since it permits testing the valves under operating conditions closer to those that would exist for an actual demand for the valves to function.

M31 An explicit CTS Applicability requirement for the ADVs lines does not exist. However, CTS 3.4.7.a and 3.4.7.b imply an Applicability for ADVs the ADV flow paths of RCS temperature ≥ 350°F since below this temperature, no further action is required for one or more inoperable ADV flow paths. ITS 3.7.4 has an applicability of MODE 1, 2, 3 and MODE 4 when SGs are relied upon for heat removal. The additional OPERABILITY requirements when in MODE 3 with RCS temperature less than 350°F and in MODE 4 when SGs are relied upon for heat removal is a more restrictive requirement upon unit operation and is consistent with the NUREG. This is appropriate since the analyses assumes ADVs are used to cool down the unit to MODE 4 for small break LOCAs.

M32 With an inoperable LPSW pump, CTS 3.3.7.b ultimately requires the unit RCS temperature be reduced to < 250°F and RCS pressure < 350 psig. The condition is comparable to ITS MODE 4. ITS 3.7.7 RA B.2 requires the unit be placed in MODE 5. This is a more restrictive requirement upon unit operation and is consistent with the NUREG. In MODES 1, 2, 3, and 4, the LPSW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the LPSW System. Therefore, the LPSW System is required to be OPERABLE in these MODES. Placing the unit in MODE 5 is necessary to exit the Applicability for the Specification.

M33 A CTS action comparable to ITS 3.7.6 Action A does not exist. With the water volume in the Condensate Storage Tank (CST), Upper Surge Tanks (UST) and condenser hotwell (HW) less than the specified limit, CTS actions are specified by CTS 3.0 which requires the unit be placed in Hot Shutdown within 12 hours and RCS temperature < 250°F within 36 hours. Since the CTS Applicability for this specification is with RCS temperature > 250°F, CTS 3.0 only requires the RCS temperature be less than 250°F within 36 hours instead of Cold Shutdown. ITS 3.7.6 RA A.2 requires the unit be in MODE 4 without reliance upon steam generators within 24 hours. The requirement to place the unit in MODE 4 without reliance upon steam generators within 24 hours is a more restrictive requirement upon unit operations and is consistent with the NUREG. Requiring the unit be placed in MODE 4 without reliance upon steam generators is appropriate since in this condition the EFW water sources are not required to be support the OPERABILITY of the EFW system.

2 M34 A CTS requirement comparable to ITS 3.7.16 does not exist. This
2 Specification adds an LCO, Applicability, Actions and Surveillance
2 Requirements for the Control Room Area Cooling System (CRACS). The
2 design basis of the CRACS is to maintain control area temperature to
2 ensure long term operability of vital equipment. The addition of ITS
2 Specification 3.7.16 is a more restrictive requirement upon unit
2 operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

L1 With one PRVS train inoperable, CTS 3.15.1.a requires daily verification of the OPERABILITY of the active components of the other train. This requirement is not retained in the ITS. Increased testing has not been shown to demonstrate operability any better than testing at the normal SR test interval. In many cases, increased testing adds to the failure rates of components by increasing wear and tear. Common mode failure analysis in conjunction with loss of safety function analyses provide adequate assurance of redundant system operability. Loss of function determination program controls are provided by ITS 5.5.16.

L2 CTS 3.4.4 requires 16 MSRVs to be OPERABLE. ITS LCO 3.7.1 requires 14 MSRVs (7 per main steam line) to be OPERABLE. A reanalysis has been performed demonstrating that ASME Code requirements are met when only 7 MSRVs are OPERABLE on each main steam line. This reanalysis was performed to determine the magnitude of power reduction and RPS overpower trip setpoint reduction required for one inoperable MSRV on each steam line necessary to establish the limits for NUREG 3.7.1 Action A and associated NUREG Figure 3.7.1-1. This reanalysis was performed for only one inoperable MSRV on each steam line based on ONS operating experience which indicated few previous MSRV failures as well as minimizing analysis costs. The analysis results indicated that the ASME Code overpressure requirements are met with one MSRV inoperable on each steam line without any power reduction or associated RPS overpower trip setpoint reduction. This change is a less restrictive requirement upon unit operation and is consistent with the NUREG.

L3 A CTS note comparable to ITS SR 3.7.1.1 Note does not exist. This Note permits entry into MODE 3 to perform the testing. Currently, this testing may be performed during the pressure and temperature reduction for a shutdown, during the refueling outage by bench testing or during the next unit startup provided the surveillance test is within its surveillance frequency. The addition of the Note for ITS SR 3.7.1.1 will allow entry into MODE 3 and testing in MODE 3 during the startup following an outage regardless of consideration of the surveillance interval. This is consistent with current practice at other nuclear power plants and is considered an acceptable method for testing of these valves. Permitting entry into MODE 3 without the SR being within its SR frequency is a less restrictive change and is consistent with NUREG 1430.

2 L4 Not used.

L5 With three EFW pumps and/or both EFW flow paths inoperable, CTS 3.4.3.d requires the initiation of immediate action to restore at least one EFW pump and associated EFW flow path to OPERABLE status. Additionally, CTS 3.4.3.d requires the unit be placed in Hot Shutdown within 12 hours and below 250°F in an additional 12 hours. The Note to ITS 3.7.5 RA E.1 permits remaining in the existing MODE until at least one EFW pump and one EFW flow path are restored to OPERABLE status. With the three EFW

pumps and/or two EFW flow paths inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with only limited means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the units into a less safe condition. The ITS 3.7.5 RA E.1 provision to suspend ITS LCO 3.0.3 and other LCO requirements for MODE changes is a less restrictive requirement upon unit operation and is consistent with the NUREG.

L6 CTS Table 4.1-3 Item 5 requires a gamma isotopic analysis be performed weekly. ITS SR 3.7.14.1 requires Secondary Specific DOSE EQUIVALENT I-131 be verified less than the specified limit on a 31 day frequency. The reduction in the surveillance frequency from weekly to every 31 days is a less restrictive requirements upon unit operations and is consistent with the NUREG. The 31 day SR Frequency is acceptable since it is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

L7 If a Main Feedwater control valve (includes both MFCVs and SFCV - refer to DOC A27) is inoperable, CTS 3.7.5.2.1.a requires the affected valve(s) to be closed. ITS 3.7.3 RA A.1 requires closing or isolating the MFCV. ITS 3.7.3 RA B.1 requires closing or isolating the SFCV. The provision to isolate the MFCV or SFCV by closing another valve in the flow path is a less restrictive requirement upon unit operation and is consistent with the NUREG. This is acceptable since isolating a MFCV or SFCV by closing another valve in the flow path is just as effective in terminating flow through the associated flow path. This change is a less restrictive requirement upon unit operation and is consistent with the NUREG.

L8 CTS 4.5.4.1.a requires each train of the PRVS be run for 15 minutes at design flow ± 10% monthly. ITS SR 3.7.10.1 requires operating each PRVS train ≥ 15 minutes at a Frequency of 31 days. The elimination of the requirement that the fans operate within 10% of design flow during this SR is a less restrictive requirement upon unit operation and is consistent with the NUREG. The purpose of this 31 day fan operation is to demonstrate the function of the system. ITS SR 3.7.10.3 adequately verifies the flow capability of the PRVS trains, although at a reduced Frequency. ITS SR 3.7.10.3, which is performed at an 18 month on a STAGGERED TEST BASIS Frequency, adequately verifies system flow meets requirements. ITS SR 3.7.10.1 adequately demonstrates system function, which is the purpose of the 31 day test. This test has typically met its flow requirements when tested at the monthly frequency.

L9 CTS 3.5.7 requires the MFCVs and SFCVs to be OPERABLE when the Reactor Coolant System is greater than 250°F. ITS 3.7.5 requires these valves to be OPERABLE in MODES 1, 2, and 3 except when all MFCVs and SFCVs are

closed and deactivated or isolated by a closed manual valve. ITS MODES 1, 2, and 3 are equivalent to CTS "...greater than 250°F." The exception "when all MFCVs and SFCVs are closed and deactivated or isolated by a closed manual valve" is acceptable since the safety function of these valves is being fulfilled when these valves are closed or isolated by a closed manual valve. This change is a less restrictive requirement upon unit operation and is consistent with the NUREG.

TECHNICAL CHANGES - REMOVAL OF DETAILS

LA1 Not used.

LA2 CTS Table 4.1-2, Note 4 includes requirements associated with the ASME Boiler and Pressure Vessel (B&PV) Code Section XI requirements for relief valve testing frequency. This information is relocated to the Inservice Testing (IST) Program. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to the Inservice Testing Program are controlled by 10 CFR 50.59 and 10 CFR 50.55a. Additionally, ITS 5.5.9 provide requirements regarding this IST program. This change is consistent with the NUREG.

LA3 CTS Table 4.1-2, Item 5 includes requirements for testing the movement capability of the TSVs. This information is relocated to the Inservice Testing (IST) Program. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to the Inservice Testing Program are controlled by 10 CFR 50.59 and 10 CFR 50.55a. This change is consistent with the NUREG.

LA4 CTS 4.9.1, 4.9.3 and 4.9.4 include details of the methodology for testing the EFW system. This information is relocated to the UFSAR Chapter 16. This information provides details of testing which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to UFSAR Chapter 16 are controlled by 10 CFR 50.59. Additionally, ITS 5.5.9 provide requirements regarding this IST program. This change is consistent with the NUREG.

LA5 CTS 3.4.1.a provides details regarding the steam supply requirements for the turbine driven EFW pump. This information is relocated to the ITS Bases. This information provides details of system operation which are

not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to ITS Bases are controlled by the Technical Specification Bases Control Program included in ITS Section 5.5. This change is consistent with the NUREG.

LA6 Not used.

LA7 CTS 4.5.1.1.2.a.2 and 4.5.1.1.2.b provide details regarding testing for specified LPSW valves. This information is relocated to the UFSAR Chapter 16. This information provides details of testing which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to UFSAR Chapter 16 are controlled by 10 CFR 50.59. Additionally, ITS 5.5.9 provide requirements regarding this IST program. This change is consistent with the NUREG.

LA8 CTS 4.12.1.a provides details regarding filter testing. This information is relocated to the ITS Bases. This information provides details of system operation which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases provides adequate assurance that they will be maintained. Changes to ITS Bases are controlled by the Technical Specification Bases Control Program included in ITS Section 5.5.

LA9 Not used.

LA10 CTS 3.19.1.a, 3.19.2.a, 3.19.1.b, 3.19.2.b, and 3.19.3 provide details regarding requirements for an OPERABLE ECCW siphon header. Footnote 6 to CTS Table 4.1-2 provides details regarding testing the Emergency Condenser Circulating Water System. This information is relocated to the ITS Bases. This information provides details of system operation which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary

to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases provides adequate assurance that they will be maintained. Changes to ITS Bases are controlled by the Technical Specification Bases Control Program included in ITS Section 5.5.

LA11 CTS 4.12.1.a includes requirements for proving all louvers in the Control Room Booster Fan Trains are OPERABLE. This information is relocated to the ITS Bases. This information provides details of system operation which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases provides adequate assurance that they will be maintained. Changes to ITS Bases are controlled by the Technical Specification Bases Control Program included in ITS Section 5.5.

LA12 CTS 3.14 and 4.18 specify requirements for snubbers. These requirements are relocated to UFSAR Chapter 16. This requirements are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to UFSAR Chapter 16 are controlled by 10 CFR 50.59. This change is consistent with the NUREG.

LA13 CTS 4.9.2 requires testing automatic valves in the EFW System in accordance with CTS 4.0.4. CTS 4.0.4 requires inservice testing of ASME Code Class 1, 2 and 3 pumps and valves be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components. This information is relocated to the Inservice Testing Program. This information provides details of testing requirements which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Inservice Testing Program is controlled by 10 CFR 50.55a. This change is consistent with the NUREG.

LA14 CTS Table 4.1-2, Item 12 and associated Note 7 provides details regarding equipment functional testing requirements for equipment actuated by the MSLB Feedwater isolation feature. These requirements are relocated to UFSAR Chapter 16. This requirements are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of testing. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. Changes to UFSAR Chapter 16 are controlled by 10 CFR 50.59. This change is consistent with the NUREG.

RELOCATED SPECIFICATIONS

R1 The CTS 3.4.6 requirements associated with the independence of the controls of the emergency feedwater system from the integrated control system is relocated to UFSAR Chapter 16. This Limiting Conditions for Operation, is not retained in the ITS because it has been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Station Unit 1, 2, and 3 Technical Specifications."

R2 CTS Table 4.1-2, Item 9 includes requirements associated with testing the spent fuel pool (SFP) cooling system. CTS does not include associated LCO requirements. This requirement is relocated to UFSAR Chapter 16. This surveillance requirement, which exists without an associated Limiting Conditions for Operation, is not retained in the ITS because it has been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Station Unit 1, 2, and 3 Technical Specifications."

R3 CTS Table 4.1-2 Item 8 includes requirements associated with testing the High Pressure Service Water Pumps and Power Supplies. CTS does not include associated LCO requirements. These requirements is relocated to UFSAR Chapter 16. These surveillance requirements, which exists without an associated Limiting Conditions for Operation, are not retained in the ITS because they have been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Station Unit 1, 2, and 3 Technical Specifications."

R4 CTS 3.8.12 and 4.14 requirements for the Reactor Building Purge and Spent Fuel Pool Ventilation System are relocated to UFSAR Chapter 16.

This Limiting Condition for Operation is not retained in the ITS because it has been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Station Unit 1, 2, and 3 Technical Specifications."

R5 CTS 3.12 requirements for the Reactor Building Polar Crane and Auxiliary Hoist are relocated to UFSAR Chapter 16. This Limiting Conditions for Operation, is not retained in the ITS because it has been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Station Unit 1, 2, and 3 Technical Specifications."

R6 Not used.

R7 CTS 4.16 includes testing requirements for Radioactive Material Sources. CTS does not include associated LCO requirements. This surveillance requirement is not retained in the ITS because the tested equipment have been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Station Unit 1, 2, and 3 Technical Specifications."

LESS RESTRICTIVE CHANGE L4

2 Not used

LESS RESTRICTIVE CHANGE L5

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> With three EFW pumps and/or both EFW flow paths inoperable, CTS 3.4.3.d requires the initiation of immediate action to restore at least one EFW pump and associated EFW flow path to OPERABLE status. Additionally, CTS 3.4.3.d requires the unit be placed in Hot Shutdown within 12 hours and below 250°F in an additional 12 hours. The Note to ITS 3.7.5 RA E.1 permits remaining in the existing MODE until at least one EFW pump and one EFW flow path are restored to OPERABLE status. With the three EFW pumps and/or two EFW flow paths inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with only limited means for conducting a cooldown. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the units into a less safe condition. The ITS 3.7.5 RA E.1 provision to suspend ITS LCO 3.0.3 and other LCO requirements for MODE changes is a less restrictive requirement upon unit operation and is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Power Company has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The change adds Required Actions for the condition where three EFW trains are inoperable. The EFW System is not assumed to be an initiator of any accident previously evaluated. The probability of occurrence of an accident is independent of the OPERABILITY of the EFW System. The accident analyses assume that an EFW System is OPERABLE to mitigate the consequences of an accident. While an inoperable EFW System may be capable of functioning to some degree, the

assumptions used in the accident analyses are not assured. This change will suspend all MODE changes until such time as one EFW pump and flow path are restored to OPERABLE status, and will require that such actions be initiated immediately upon entering the Condition. With an inoperable EFW System, the unit would be in a seriously degraded condition, and should not be perturbed by any action, including a power change, that might result in a trip. Any transient could force the unit into a less safe condition. The proposed change will assure that an EFW pump and flow path are OPERABLE prior to initiating any unit transient or shutdown operation. Therefore, this change does not involve a significant increase in the probability of occurrence, or a significant increase the consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The addition of Required Actions to suspend required MODE changes of LCO 3.0.3 does not affect any new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed change provides Required Actions for the plant condition where the entire EFW System is inoperable. This change requires restoration of at least one EFW train to OPERABLE status before permitting a MODE change. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L6

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 4.1-3 Item 5 requires a gamma isotopic analysis be performed weekly. ITS SR 3.7.14.1 requires Secondary Specific DOSE EQUIVALENT I-131 be verified less than the specified limit on a 31 day frequency. The reduction in the surveillance frequency from weekly to every 31 days is a less restrictive requirements upon unit operations and is consistent with the NUREG. The 31 day SR Frequency is acceptable since it is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Power Company has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. This change effectively reduces the Surveillance Frequency for secondary coolant activity limits from 7 days to 31 days. The Surveillance Frequency is not assumed to be an initiator of any accident previously evaluated. The probability of occurrence of an accident is independent of the Surveillance Frequency for activity in the secondary coolant. The 31 day Frequency provides adequate assurance that adverse trends of the level of iodine in the secondary coolant would be detected and allow for appropriate action to maintain the levels within specified limits. The consequences of a design basis accident are not affected by the Surveillance Frequency. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The Surveillance Frequency for secondary coolant activity does not affect any new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

This change involves reducing the Surveillance Frequency for secondary coolant activity from 7 days to 31 days. The 31 day Frequency provides adequate assurance that adverse trends of the level of iodine in the secondary coolant would be detected and allow for appropriate action to maintain the levels within specified limits. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L7

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> If a Main Feedwater control valve (includes both MFCVs and SFCV - refer to DOC A27) is inoperable, CTS 3.7.5.2.1.a requires the affected valve(s) to be closed. ITS 3.7.3 RA A.1 requires closing or isolating the MFCV. ITS 3.7.3 RA B.1 requires closing or isolating the SFCV. The provision to isolate the MFCV or SFCV by closing another valve in the flow path is a less restrictive requirement upon unit operation and is consistent with the NUREG. This is acceptable since isolating a MFCV or SFCV by closing another valve in the flow path is just as effective in terminating flow through the associated flow path. This change is a less restrictive requirement upon unit operation and is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Power Company has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The means of isolating a feedwater line with an inoperable MFCV or SFCV is not assumed to be an initiator of any accident previously evaluated. The probability of occurrence of an accident is therefore independent of the manner in which the isolation is accomplished. This change provides the option to isolate flow through the associated feedwater line in an equally effective manner. Since the change still requires isolation of the line, albeit in a different manner, the consequences of a design basis accident are not affected by the change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. Isolation the line is still required, although in a different way. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. **Does this change involve a significant reduction in a margin of safety?**

This change involves permitting isolation of a feedwater line with an inoperable MFCV or SFCV in an different, but equally effective, manner. Since the change still requires the line to be isolated, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L8

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 4.5.4.1.a requires each train of the PRVS be run for 15 minutes at design flow ± 10% monthly. ITS SR 3.7.10.1 requires operating each PRVS train ≥ 15 minutes at a Frequency of 31 days. The elimination of the requirement that the fans operate within 10% of design flow during this SR is a less restrictive requirement upon unit operation and is consistent with the NUREG. The purpose of this 31 day fan operation is to demonstrate the function of the system. ITS SR 3.7.10.3 adequately verifies the flow capability of the PRVS trains, although at a reduced Frequency. ITS SR 3.7.10.3, which is performed at an 18 month on a STAGGERED TEST BASIS Frequency, adequately verifies system flow meets requirements. ITS SR 3.7.10.1 adequately demonstrates system function, which is the purpose of the 31 day test. This test has typically met its flow requirements when tested at the monthly frequency.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Power Company has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The frequency for verifying the flow of the PRVS is not assumed to be an initiator of any accident previously evaluated. The probability of occurrence of an accident is therefore independent of the Frequency of the flow verification. The Surveillance Frequency for the PRVS flow verification does not serve an accident mitigation function. The LCO still requires the PRVS to be OPERABLE. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. The PRVS flow is still verified although on a lower Frequency. The PRVS is still required to be OPERABLE to serve its accident mitigation function. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. **Does this change involve a significant reduction in a margin of safety?**

This change involves eliminating the verification of PRVS system flow during the monthly operation of the trains. PRVS system flow is still required to be verified by ITS SR 3.7.10.4 although at a reduced frequency. This test has typically met its flow requirements when tested at the monthly frequency. Verification of the flow during performance of SR 3.7.10.4 provides adequate assurance the PRVS is capable of performing its intended function. Therefore this change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT

This proposed Technical Specification Change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22 (c) (9). The following is a discussion of how the proposed Technical Specification Change meets the criteria for categorical exclusion.

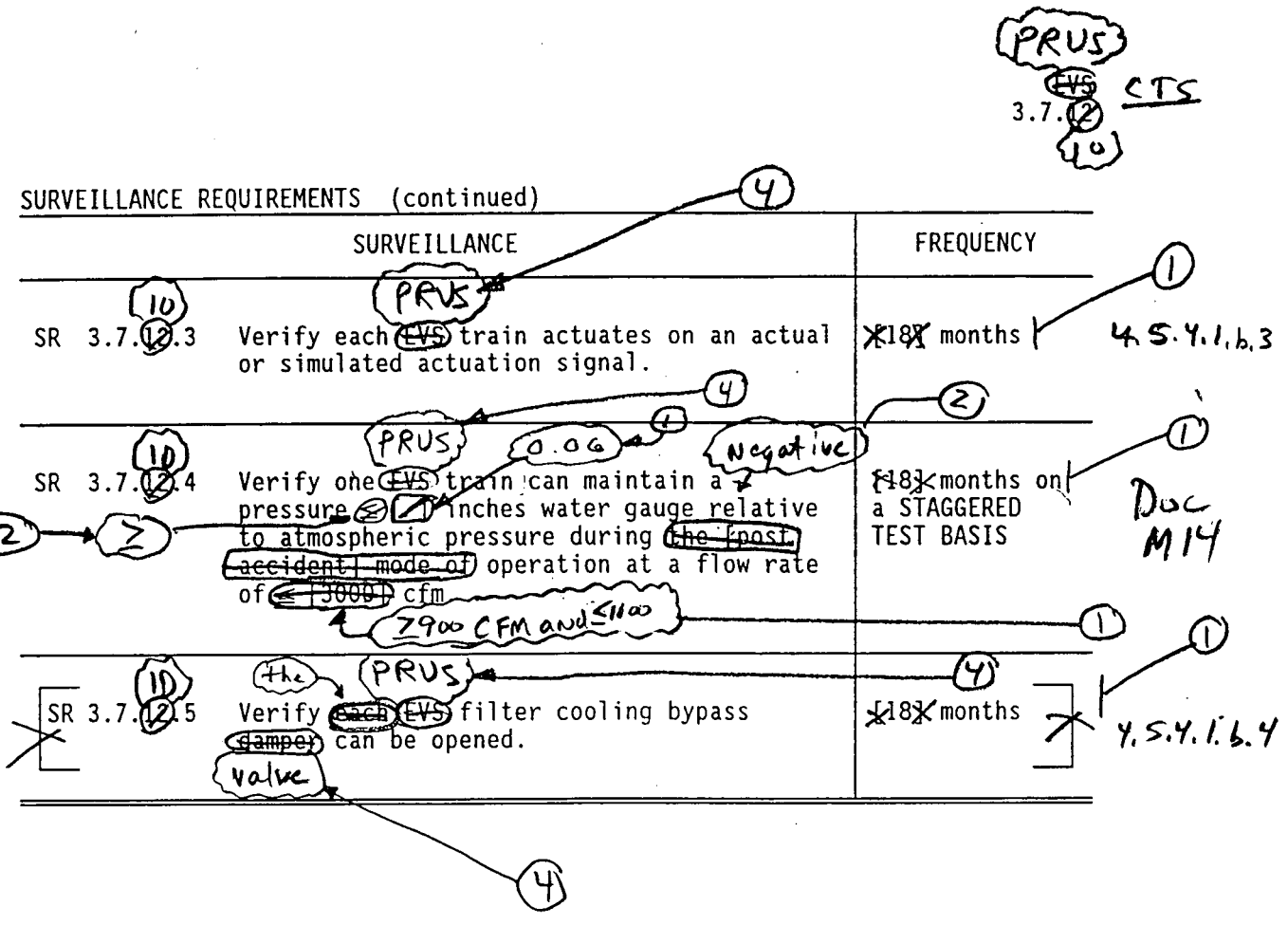
10 CFR 51.22 (c) (9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements;

(i) the proposed change involves no Significant Hazards Consideration (refer to the No Significant Hazards Consideration section of this Technical Specification Change Request),

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths, and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed changes of this request.



SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.~~12~~.3	Verify each ~~EVS~~ train actuates on an actual or simulated actuation signal.	~~18~~ months
SR 3.7.~~12~~.4	Verify one ~~EVS~~ train can maintain a pressure ~~≤~~ [ ] inches water gauge relative to atmospheric pressure during ~~the [post-accident] mode of~~ operation at a flow rate of ~~≤ [3000]~~ cfm	~~18~~ months on a STAGGERED TEST BASIS
SR 3.7.~~12~~.5	Verify ~~each~~ ~~EVS~~ filter cooling bypass ~~damper~~ valve can be opened.	~~18~~ months

BACKGROUND

The ~~EVS~~ filters air from the area of the active ~~emergency~~ ~~Core Cooling System (ECCS) components~~ during the recirculation phase of a loss of coolant accident (LOCA).

The ~~EVS~~ consists of two independent, redundant trains. Each train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated ~~charcoa~~ adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the ~~Auxiliary~~ Building ~~negative pressure~~ area following receipt of ~~a safety features~~ actuation signal (an Engineered Safeguards ESAS ~~(SFAS)~~).

The ~~EVS~~ is a standby system. During emergency operations, the ~~EVS~~ ~~dampers~~ are realigned, and fans are started to begin filtration. Upon receipt of the ~~SFAS~~ signal(s), ~~normal air discharges from the negative pressure area are isolated, and~~ the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and ~~charcoa~~ adsorbers.

The ~~EVS~~ is discussed in the FSAR, Sections ~~6.2.3~~, ~~9.4.2~~, 9.4.7 and ~~15.4.6~~ 15.4.7 (Refs. 1, 2, and 3, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the ~~EVS~~ is established by the ~~large break LOCA~~. The system evaluation assumes a passive failure of the ECCS outside containment, such as an ECCS pump seal failure during the recirculation mode. In such a case, the system limits radioactive release to within 10 CFR 100 (Ref. ~~4~~) requirements. The analysis of the effects and consequences of a ~~large break LOCA~~ is presented in Reference 3. The ~~EVS~~ also actuates following a ~~small~~ break LOCA, in those cases where the unit goes into the recirculation mode of long term cooling, and to cleanup releases of smaller leaks, such as from valve stem packing.

Supp 2 | and personnel doses in the Control Room are maintained within the limits of 10CFR20 (Ref. 4). No credit is taken in the analysis for any reduction in Control Room Dose provided by the PRUS Filters.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 6.2.3 → 6.5.1
2. FSAR, Section 9.4.2 → 9.4.7
3. FSAR, Section 15.4.6 → 15.4.7
4. 10 CFR 100.1 → 20
5. Regulatory Guide 1.52.
6. 7. 10 CFR 100.

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5. 10 CFR 50.36.

ITS Section 3.7

ID 140

Subject: Revised ITS to add NUREG Specification on Control Area Cooling.

3.7 PLANT SYSTEMS

3.7.16 Control Room Area Cooling Systems (CRACS)

LCO 3.7.16 Two CRACS trains shall be OPERABLE as follows:

a. Two trains of the Control Room Ventilation System (CRVS) shall be OPERABLE, and

b. Two trains of the Chilled Water (WC) System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRVS train inoperable.	A.1 Restore CRVS train to OPERABLE status.	30 days
B. One WC train inoperable.	B.1 Restore WC train to OPERABLE status.	30 days
C. Control Room area air temperature not within limit.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	C.1 Restore Control Room area air temperature within limit.	7 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	12 hours 36 hours
E. Two CRVS trains inoperable. OR Two WC Trains inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify temperature in Control Room and Cable Room is $\leq$ 80°F and temperature in Electrical Equipment Room is $\leq$ 85°F.	12 hours

B 3.7 PLANT SYSTEMS

B 3.7.16 Control Room Area Cooling Systems (CRACS)

BASES

BACKGROUND

The CRACS provides temperature control for the control areas.

The control area is defined as the control room, cable room, and equipment room for each unit. Units 1 and 2 have a shared control room, and Unit 3 has an independent control room. The cable and equipment rooms are independent for each unit. The control rooms, cable rooms, and equipment rooms for each unit contain vital electrical equipment, such as 125 VDC Vital I&C Power and 120 VAC Vital I&C Power, which is essential for achieving safe shutdown on the units. A control area portion is defined as a cable room, equipment room, or control room, for which a set of redundant CRVS cooling trains is required. The control area portions are listed in the table below. Through the use of alternative air flow paths, air handling units AHU-34 and AHU-35 provide redundant cooling to both Units 1 and 2 cable rooms.

The AHUs which cool the control areas are part of the CRVS for each unit. The Chilled Water System (WC) serves as the heat sink for the CRVS on all three units. The WC System consists of two redundant cooling trains which serve all three units.

UFSAR Section 3.11.4 (Ref. 1) requires that redundant air conditioning and ventilation equipment be available to assure that no single failure of an active component within the CRVS and WC System will prevent proper control area environmental control. During a LOOP event, power will be temporarily lost to the equipment within these systems. Upon restoration of power the equipment will be required to restart. This restart makes the equipment susceptible to a single active failure. Without redundant cooling capability, acceptable temperatures within the control area could be exceeded. This could result in the potential failure of vital electrical equipment which is needed for safe shutdown of the units.

(continued)

BASES

BACKGROUND (continued)

The following table identifies each portion of the CRVS where redundancy is required:

Table B 3.7.16-1
CRVS Redundant Equipment

Control Area Portion	Associated CRVS Cooling Trains
Unit 1&2 Control Room	AHU-11 and AHU-12
Unit 1 Cable Room	AHU-34 and AHU-35
Unit 1 Equipment Room	AHU-22 and AHU-34
Unit 2 Cable Room	AHU-34 and AHU-35
Unit 2 Equipment Room	AHU-23 and AHU-35
Unit 3 Control Room	AHUs 3-13 and 3-14
Unit 3 Cable Room	AHUs 3-11 and 3-12
Unit 3 Equipment Room	AHUs 3-15 and 3-16

A single train will provide the required temperature control. The CRACS operation to maintain control room temperature is discussed in the UFSAR, Section 9.4.1 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The design basis of the CRACS is to maintain control area temperature to ensure cooling of vital equipment.

The CRACS components are arranged in redundant trains. A single active failure of a CRACS component does not impair the ability of the system to perform as designed. The CRACS is designed to remove sensible and latent heat loads from the control area, including consideration of equipment heat loads to ensure equipment OPERABILITY.

The CRACS satisfies Criterion 3 of the NRC Policy Statement.

BASES (continued)

LCO

Two redundant trains of the CRACS and WC Systems train are required to be OPERABLE to ensure that at least one train in each system is available, assuming a single active failure disables the other train in one or both systems. Total system failure could result in the equipment operating temperature exceeding limits. A Train of CRVS consists of one of the redundant AHUs specified in Table B 3.7.16-1 for each of the three portions of the control area for an Oconee unit and associated ducts, dampers, instrumentation and controls. A single AHU can function as a component in more than one train on an Oconee unit and can function as a component on trains in multiple Oconee units. For example AHU-34, and its associated ducts, damper, instrumentation and controls, can simultaneously function as the AHU for a train of CRVS serving the Unit 1 cable room, the Unit 1 equipment room as well as the Unit 2 cable room. The combination of AHU-34 and either AHU-11 or AHU-12 along with their associated equipment constitutes a combination of equipment which can satisfy the requirement for one train of CRVS for Unit 1. Additionally, AHU-34 can simultaneously serve as the AHU for the portion of a Unit 2 CRVS train serving the Unit 2 cable room. AHU-35 in combination with either AHU-11 or AHU-12 along with their associated equipment constitutes a combination of equipment which can satisfy the requirement for one train of CRVS for Unit 2.

The CRACS is considered OPERABLE when the individual components that are necessary to maintain control area temperature are OPERABLE in both trains of CRVS and WC System. These components include the cooling coils, water cooled condensing units, and associated temperature control instrumentation. The two redundant trains can include a temporarily installed full-capacity control area cooling train. Any temporary cooling train shall have a power source with availability equivalent to the source of the permanently installed train. In addition, the CRACS must be OPERABLE to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4 the CRACS must be OPERABLE to ensure that the control area temperature will not exceed equipment OPERABILITY requirements.

(continued)

BASES (continued)

ACTIONS

A.1

With one CRVS train inoperable for the control area, action must be taken to restore the CRVS train to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CRVS train is adequate to maintain the control area temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE CRVS train could result in a loss of CRVS cooling function. The 30 day Completion Time is based on the low probability of a loss of CRVS cooling component and the time necessary to perform repairs to CRVS cooling equipment.

B.1

With one WC train inoperable for a control area portion, action must be taken to restore the WC train to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE WC train is adequate to maintain the control area portion temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE WC train could result in a loss of CRACS cooling function. The 30 day Completion Time is based on the low probability of a loss of WC cooling component, and on the time necessary to perform repairs to WC cooling equipment.

C.1

With the control room area air temperature outside its limit, action must be taken to restore the air temperature to within the limit within 7 days. If the control room area air temperature exceeds its limit, the ability of a single train of CRACS to maintain control room area temperature may be affected. The completion time of 7 days is reasonable considering the remaining CRACS train available to perform the required temperature control function and the low probability of an event occurring that would require the CRACS operation during that time.

The Required Actions are modified by a Note that states LCO 3.0.4 is not applicable. In consideration of the redundant CRACS train available, the small variation in temperature expected between 12 hour surveillances, and the marginal impact small temperature variations may have on the ability of a CRACS train to maintain the control room

(continued)

BASES (continued)

ACTIONS

C.1 (continued)

temperature within limits, an exception to LCO 3.0.4 is applicable for this condition.

D.1 and D.2

If the Required Actions and associated Completion Times of Conditions A, B, or C are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

E.1

If both CRVS trains or both WC trains are inoperable, the CRACS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies that the heat removal capability of the system is sufficient to maintain the temperature in the control room and cable room at or below 80°F and maintain the temperature in the electrical equipment room at or below 85°F. The temperature is determined by reading gauges in each area which are considered representative of the average area temperature. These temperature limits are based on operating history and are intended to provide an indication of degradation of the cooling systems. The limits are conservative with respect to equipment operability temperature limits. The values for the SR are values at which the system is removing sufficient heat to meet design requirements (i.e., OPERABLE) and sufficiently above the values associated with normal operation during hot weather. The temperature in the equipment room is typically slightly higher than the temperature in the control room or cable room. Because of that, a higher value is specified for this

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1 (continued)

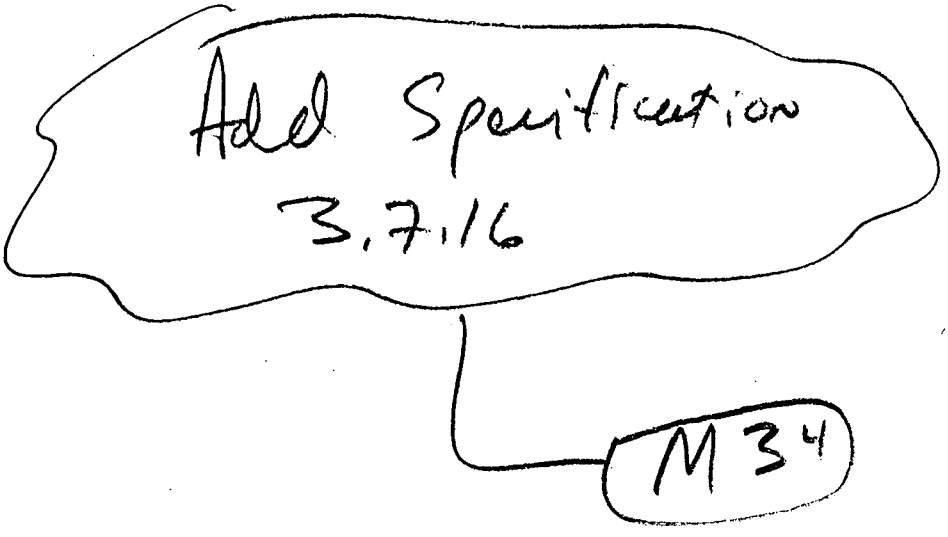
area. The 12 hour Frequency is appropriate since significant degradation of the CRACS is slow and is not expected over this time period.

REFERENCES

1. UFSAR, Section 3.11.4.
2. UFSAR, Section 9.4.1.

Specification
3.7.16

Supp.
2



M32 With an inoperable LPSW pump, CTS 3.3.7.b ultimately requires the unit RCS temperature be reduced to < 250°F and RCS pressure < 350 psig. The condition is comparable to ITS MODE 4. ITS 3.7.7 RA B.2 requires the unit be placed in MODE 5. This is a more restrictive requirement upon unit operation and is consistent with the NUREG. In MODES 1, 2, 3, and 4, the LPSW System is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the LPSW System. Therefore, the LPSW System is required to be OPERABLE in these MODES. Placing the unit in MODE 5 is necessary to exit the Applicability for the Specification.

M33 A CTS action comparable to ITS 3.7.6 Action A does not exist. With the water volume in the Condensate Storage Tank (CST), Upper Surge Tanks (UST) and condenser hotwell (HW) less than the specified limit, CTS actions are specified by CTS 3.0 which requires the unit be placed in Hot Shutdown within 12 hours and RCS temperature < 250°F within 36 hours. Since the CTS Applicability for this specification is with RCS temperature > 250°F, CTS 3.0 only requires the RCS temperature be less than 250°F within 36 hours instead of Cold Shutdown. ITS 3.7.6 RA A.2 requires the unit be in MODE 4 without reliance upon steam generators within 24 hours. The requirement to place the unit in MODE 4 without reliance upon steam generators within 24 hours is a more restrictive requirement upon unit operations and is consistent with the NUREG. Requiring the unit be placed in MODE 4 without reliance upon steam generators is appropriate since in this condition the EFW water sources are not required to be support the OPERABILITY of the EFW system.

2 M34 A CTS requirement comparable to ITS 3.7.16 does not exist. This
2 Specification adds an LCO, Applicability, Actions and Surveillance
2 Requirements for the Control Room Area Cooling System (CRACS). The
2 design basis of the CRACS is to maintain control area temperature to
2 ensure long term operability of vital equipment. The addition of ITS
2 Specification 3.7.16 is a more restrictive requirement upon unit
2 operation.

3.7 PLANT SYSTEMS

3.7.1 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.1 Two CREATCS trains shall be OPERABLE as follows:
a. Two trains of the Control Room Ventilation System (CRVS) shall be OPERABLE, and
b. Two trains of the Chilled Water (WC) system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4, [5, and 6,]
[During movement of irradiated fuel assemblies,]
[During CORE ALTERATIONS].

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREATCS train inoperable.	A.1 Restore CREATCS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS].	C.1 Place OPERABLE CREATCS train in operation. OR C.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

(continued)

INSERT 3.7-26A

2 Insert 3.7-26A
2

2 B.	One WC train inoperable.	B.1 Restore WC train to OPERABLE status.	30 days
2 C.	Control Room area air temperature not within limit.	----------NOTE---------- LCO 3.0.4 is not applicable. ---------- C.1 Restore Control Room area air temperature within limit.	7 days

65

68

Supp. 2

(4) CRACS → CREATCS 3.7 (11) (16)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two CREATCS trains inoperable during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS].	D.1 Suspend movement of irradiated fuel assemblies.	Immediately (66)
(E) Two CREATCS trains inoperable during MODE 1, 2, 3, or 4.	(E) E.1 Enter LCO 3.0.3.	Immediately Doc M34

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify each CREATCS train has the capability to remove the assumed heat load.	[18] months Doc M34

Insert SR 3.7.16.1 (68)

Two CRUS trains inoperable. OR Two WC Trains inoperable. (67) Doc M34

2 **Insert 3.7-27A**
2

SR 3.7.16.1	Verify temperature in Control Room and Cable Room is ≤ 80°F and temperature in Electrical Equipment Room is ≤ 85°F.	12 hours

second entry Condition for NUREG 3.7.9 Action B results in entry into LCO 3.0.3 which is comparable to CTS 3.0.

62 The brackets are removed and the bracketed NUREG value of 0.10 μCi/gm adopted. This is the value for secondary specific activity assumed in the UFSAR Chapter 15 reanalysis. (The methodology for this reanalysis has been submitted separately for NRC review. Refer to submittal letter for additional information.)

63 Specification 3.7.15 is added to the NUREG to capture CTS requirements regarding decay time restrictions in specified rows in the SFPs prior to movement of casks in the SFP. This change is necessary since the ONS design requires placing a cask in the SFP, in proximity to spent fuel stored in the SFP.

64 The brackets are removed and the bracketed NUREG value of 90°F adopted. This is the value for service water inlet temperature assumed in the accident and transient.

2 65 NUREG LCO 3.7.11 is modified to specify the Control Room Area Cooling
2 System (CRACS) shall be OPERABLE with two trains of Control Room
2 Ventilation System OPERABLE and two trains of Chilled Water (WC) System
2 OPERABLE. This ITS presentation is necessary to completely identify
2 applicable requirements. The ONS design for Control Room Cooling
2 consists of two trains in each of two different systems. The Control
2 Room Ventilation System (CRVS) includes the air handling units, ducts,
2 dampers, heat exchangers, instrumentation and controls, etc. associated
2 with the ventilation system air flow. The Chilled Water (WC) System
2 includes the pumps, valves, piping, chillers, instrumentation and
2 controls, etc. associated with the chilled water supply to the CRVS heat
2 exchangers. Each train of the CRVS system can be supplied from either
2 train of the WC System.
2
2 NUREG 3.7.11 Action A is modified to provide an Action for one
2 inoperable CRVS train. A new Action B is added to provide an Action for
2 one inoperable WC train. This ITS presentation is necessary to provide
2 appropriate actions for one inoperable train in the CRVS System or the
2 WC system.
2
2 66 The bracketed statements within the Applicability statement for NUREG
2 Specification 3.7.16 and bracketed NUREG Actions C and D which involve
2 not meeting the LCO when in these bracketed MODES are not adopted.
2 There is no current licensing basis regarding the Applicability for this
2 Specification. The requirements for the CRACS are currently
2 administratively controlled when above Cold Shutdown which is comparable
2 to ITS MODES 1, 2, 3 and 4.
2
2

2 67 The entry Condition for NUREG 3.7.11 Action E is modified to include two
2 trains of CRVS inoperable or two trains of WC inoperable. Loss of two
2 trains of either of these systems involves a loss of safety function.
2 This representation is equivalent to the NUREG 3.7.11 Action E but is
2 modified to include both the CRVS and WC Systems. This is appropriate
2 because the ONS design for Control Room Cooling consists of two trains
2 in each of two different systems.
2
2 68 NUREG SR 3.7.11.1 is not adopted. This SR is not required by the
2 current licensing basis and is overly burdensome to perform. The ONS
2 CRVS is used to maintain the control area environmental conditions
2 within limits at all times. The heat load in the control area is not
2 significantly different under accident conditions than under normal
2 operating conditions. NUREG SR 3.7.11.1 is replaced with a requirement
2 to verify the control area temperatures are within specified values at a
2 12 hour Frequency. This SR is simple to perform, is performed more
2 frequently and adequately verifies the OPERABILITY of the CRVS without
2 the hardship created by requiring performance of NUREG SR 3.7.11.1.
2
2 ITS ACTION C is added to address control room air temperature not within
2 limit. This is consistent with the NUREG format to provide appropriate
2 Required Actions that correspond to the surveillances used to verify
2 system OPERABILITY. The lack of an appropriate ACTION for this
2 surveillance could, in the event this surveillance was not met, result
2 in an unnecessary entry into LCO 3.0.3 and an inappropriate plant
2 shutdown, because an ACTION is not provided corresponding to the manner
2 in which the LCO is not met. The proposed ACTION allows 7 days to
2 restore the control room air temperature to within its limit. The
2 ACTION also contains an exception to LCO 3.0.4. The time allowed to
2 restore the temperature and the exception to LCO 3.0.4 are reasonable
2 considering (a) the remaining CRACS train available to perform the
2 cooling function in an emergency, (b) the small expected deviation above
2 the limit before discovery by performance of the temperature
2 verification of ITS SR 3.7.16.1 every 12 hours, and C the marginal
2 impact that a control room temperature slightly above the limit would
2 have on the ability of CRACS to perform its emergency cooling function.
2 Therefore, the proposed ACTION is acceptable.
2
2 ITS ACTION and SR are similar to that approved for other plants
2 (Vogtle).

BACKGROUND The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment. Ductwork, valves or dampers, and instrumentation also form part of the system. Two redundant air cooled condensing units are provided as a backup to the water cooled condensing unit. Both the water cooled and air cooled condensing units must be OPERABLE for the CREATCS to be OPERABLE. During emergency operation, the CREATCS maintains the temperature between 70°F and 85°F. The CREATCS is a subsystem providing air temperature control for the control room.

The CREATCS is an emergency system. On detection of high containment building pressure or radiation, low Reactor Coolant System pressure, or high noble gas radioactivity in the station vent, the normal control room ventilation system is automatically shut down, and the Control Room Emergency Ventilation System can be manually started. A single train will provide the required temperature control. The CREATCS operation to maintain control room temperature is discussed in the FSAR, Section 9.4.1 (Ref. 1, 2).

APPLICABLE SAFETY ANALYSES The design basis of the CREATCS is to maintain control room temperature for 30 days of continuous occupancy.

The CREATCS components are arranged in redundant, safety related trains. During emergency operation, the CREATCS maintains the temperature between [70]°F and [95]°F. A single active failure of a CREATCS component does not impair the ability of the system to perform as designed. The CREATCS is designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent heat loads from the control room, including

(continued)

2 Insert B3.7-55A

2 The control area is defined as the control room, cable room, and equipment
2 room for each unit. Units 1 and 2 have a shared control room, and Unit 3 has
2 an independent control room. The cable and equipment rooms are independent
2 for each unit. The control rooms, cable rooms, and equipment rooms for each
2 unit contain vital electrical equipment, such as 125 VDC Vital I&C Power and
2 120 VAC Vital I&C Power, which is essential for achieving safe shutdown on the
2 units. A control area portion is defined as a cable room, equipment room, or
2 control room, for which a set of redundant CRVS cooling trains is required.
2 The control area portions are listed in the table below. Through the use of
2 alternative air flow paths, air handling units AHU-34 and AHU-35 provide
2 redundant cooling to both Units 1 and 2 cable rooms.
2
2 The AHUs which cool the control areas are part of the CRVS for each unit. The
2 Chilled Water System (WC) serves as the heat sink for the CRVS on all three
2 units. The WC System consists of two redundant cooling trains which serve all
2 three units.
2
2 UFSAR Section 3.11.4 (Ref. 1) requires that redundant air conditioning and
2 ventilation equipment be available to assure that no single failure of an
2 active component within the CRVS and WC System will prevent proper control
2 area environmental control. During a LOOP event, power will be temporarily
2 lost to the equipment within these systems. Upon restoration of power the
2 equipment will be required to restart. This restart makes the equipment
2 susceptible to a single active failure. Without redundant cooling capability,
2 acceptable temperatures within the control area could be exceeded. This could
2 result in the potential failure of vital electrical equipment which is needed
2 for safe shutdown of the units.
2
2 The following table identifies each portion of the CRVS where redundancy is
2 required:
2

Control Area Portion	Associated CRVS Cooling Trains
Unit 1&2 Control Room	AHU-11 and AHU-12
Unit 1 Cable Room	AHU-34 and AHU-35
Unit 1 Equipment Room	AHU-22 and AHU-34
Unit 2 Cable Room	AHU-34 and AHU-35
Unit 2 Equipment Room	AHU-23 and AHU-35
Unit 3 Control Room	AHUs 3-13 and 3-14
Unit 3 Cable Room	AHUs 3-11 and 3-12
Unit 3 Equipment Room	AHUs 3-15 and 3-16

BASES

APPLICABLE SAFETY ANALYSES (continued)

consideration of equipment heat loads ~~and personnel~~ occupancy requirements, to ensure equipment OPERABILITY.

The ~~CREATCS~~ CRACS satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

The ~~CREATCS~~ CRACS is considered OPERABLE when the individual components that are necessary to maintain control ~~room~~ area temperature are OPERABLE in both trains. These components include the cooling coils, water cooled condensing units, and associated temperature control instrumentation. In addition, the ~~CREATCS~~ CRACS must be OPERABLE to the extent that air circulation can be maintained.

INSERT B3.7-56A

INSERT B3.7-56B

APPLICABILITY

In MODES 1, 2, 3, 4, ~~[5, and 6,]~~ and during movement of ~~irradiated fuel assemblies and during CORE ALTERATIONS,~~ the ~~CREATCS~~ CRACS must be OPERABLE to ensure that the control ~~room~~ area temperature will not exceed equipment OPERABILITY requirements ~~following isolation of the control room~~.

ACTIONS

A.1

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

a loss of CRUS cooling component + the time necessary to perform repairs to CRUS cooling equipment

(continued)

2 Insert B 3.7-56A

2 A Train of CRVS consists of one of the redundant AHUs specified in Table
2 B 3.7.16-1 for each of the three portions of the control area for an Oconee
2 unit and associated ducts, dampers, instrumentation and controls. A single
2 AHU can function as a component in more than one train on an Oconee unit and
2 can function as a component on trains in multiple Oconee units. For example
2 AHU-34, and its associated ducts, damper, instrumentation and controls, can
2 simultaneously function as the AHU for a train of CRVS serving the Unit 1
2 cable room, the Unit 1 equipment room as well as the Unit 2 cable room. The
2 combination of AHU-34 and either AHU-11 or AHU-12 along with their associated
2 equipment constitutes a combination of equipment which can satisfy the
2 requirement for one train of CRVS for Unit 1. Additionally, AHU-34 can
2 simultaneously serve as the AHU for the portion of a Unit 2 CRVS train serving
2 the Unit 2 cable room. AHU-35 in combination with either AHU-11 or AHU-12
2 along with their associated equipment constitutes a combination of equipment
2 which can satisfy the requirement for one train of CRVS for Unit 2.

2 Insert B 3.7-56B

2 The two redundant trains can include a temporarily installed full-capacity
2 control area cooling train. Any temporary cooling train shall have a power
2 source with availability equivalent to the source of the permanently installed
2 train.

Suppl. 2

BASES

ACTIONS

A.1 (continued)

Concurrent failure of two CREATCS trains would result in the loss of function capability; therefore, LCO 3.0.3 must be entered immediately.

If the Required Actions and associated Completion Times of Conditions A, B, or C are not met,

~~D~~ ~~B~~.1 and ~~D~~ ~~B~~.2

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

INSERT B37-57A

C.1 and C.2

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

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

D.1

[In MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS], with two CREATCS trains inoperable, action must be taken to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places

(continued)

2 **Insert B 3.7-57A**

2 B.1

2 With one WC train inoperable for a control area portion, action must be taken
2 to restore the WC train to OPERABLE status within 30 days. In this Condition,
2 the remaining OPERABLE WC train is adequate to maintain the control area
2 portion temperature within limits. However, the overall reliability is
2 reduced because a failure in the OPERABLE WC train could result in a loss of
2 CRACS cooling function. The 30 day Completion Time is based on the low
2 probability of a loss of WC cooling component, and on the time necessary to
2 perform repairs to WC cooling equipment.

2 C.1

2 With the control room area air temperature outside its limit, action must be
2 taken to restore the air temperature to within the limit within 7 days. If
2 the control room area air temperature exceeds its limit, the ability of a
2 single train of CRACS to maintain control room area temperature may be
2 affected. The completion time of 7 days is reasonable considering the
2 remaining CRACS train available to perform the required temperature control
2 function and the low probability of an event occurring that would require the
2 CRACS operation during that time.

2 The Required Actions are modified by a Note that states LCO 3.0.4 is not
2 applicable. In consideration of the redundant CRACS train available, the
2 small variation in temperature expected between 12 hour surveillances, and the
2 marginal impact small temperature variations may have on the ability of a
2 CRACS train to maintain the control room temperature within limits, an
2 exception to LCO 3.0.4 is applicable for this condition.

Supp. 2

CRACS
~~CREATCS~~
B 3.7 (M)
(4) (16)

BASES

ACTIONS

D.1 (continued)

(5)

~~the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.~~

E.1 CRUS or both MC trains

If both ~~CREATC~~ trains are inoperable ~~in MODE 1, 2, 3, or 4~~, the ~~CREATCS~~ may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

CRACS

SURVEILLANCE REQUIREMENTS

SR 3.7.~~12~~.1

(16)

~~This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the [safety analyses]. This SR consists of a combination of testing and calculations. An [18] month Frequency is appropriate, as significant degradation of the CREATCS is slow and is not expected over this time period.~~

REFERENCES

1. UFSAR, Section 3.11.4 (4)

2. ~~1.~~ UFSAR, Section ~~9.4~~ 9.4.1 (1)

Insert B3.7-58A

(5)

2 **Insert B 3.7-58A**

2 This SR verifies that the heat removal capability of the system is sufficient
2 to maintain the temperature in the control room and cable room at or below
2 80°F and maintain the temperature in the electrical equipment room at or below
2 85°F. The temperature is determined by reading gauges in each area which are
2 considered representative of the average area temperature. These temperature
2 limits are based on operating history and are intended to provide an
2 indication of degradation of the cooling systems. The limits are conservative
2 with respect to equipment operability temperature limits. The values for the
2 SR are values at which the system is removing sufficient heat to meet design
2 requirements (i.e., OPERABLE) and sufficiently above the values associated
2 with normal operation during hot weather. The temperature in the equipment
2 room is typically slightly higher than the temperature in the control room or
2 cable room. Because of this, a higher value is specified for this area. The
2 12 hour Frequency is appropriate since significant degradation of the CRACS is
2 slow and is not expected over this time period.

ITS Section 3.3
3.7

ID 142

Subject: Incorporated latest revision to TSC 95-03 - MSLB

A1 (except as marked)

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

Applicability

A27

Applies to main steam line break (MSLB) detection and feedwater isolation circuitry when main steam header pressure is greater than 700 psig and to the two Main Feedwater main and startup control (Main Feedwater control) valves when Reactor Coolant System temperature is greater than 250 °F. (SEE 3.3)

Applic

MODES 1, 2, +3 except when all MFCVs and SFCVs are closed and deactivated or isolated by a closed manual valve L9

Suppl

Objective

To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside containment.

Specifications

Two MFCVs and two SFCVs A27

3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items 20, 21, and 22. (SEE 3.3)

3.5.7.2 The Main Feedwater control valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

L7

RA A.1, A.2 RA B.1, B.2

a. ~~A~~ One Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed or isolated within 8 hours from discovery and verified closed once per 7 days. A28

RA C.1, C.2

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a hot shutdown condition (MODE 3) within 12 hours, and be less than or equal to an RCS temperature of 250 °F (MODE 4) in an additional 18 hours. M18

Add ACTION Note A20

Add SR 3.7.3.1 M30

Bases

The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner. A2

Supp 2

Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

A1 <except as marked>

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	<SEE 3.3>
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(I)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide Range	ES(1)	PS	NA	(1) When in service.
SR 3.9.2.1 6. Source Range	~~ES(1)~~ A5 → 12 hrs	PS	~~NA~~	~~(1) When in service.~~ MODE 6 – M5
7. Reactor Coolant Temperature	ES	45 Days STB	18 months	Add SR 3.9.2.2 – M7
8. High Reactor Coolant Pressure	ES	45 Days STB	18 months	<SEE 3.3>
9. Low Reactor Coolant Pressure	ES	45 Days STB	18 months	

3.9.3 supp 2

Specification 3.9.2

Amendment No. 228 (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

ITS Section 3.9

ID 112

Subject: Section 3.9, DOC A6 inserted and characterized as "not used." Section 3.9, DOC A8 - change "on" to "in."

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with NUREG-1430, Revision 1. As a result, the Technical Specifications (TS) should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430, Revision 1. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1430, Revision 1. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A2 The ONS 1, 2, & 3 CTS Bases are completely replaced by revised bases that reflect the format and applicable content of proposed ITS Section 3.9. The revised Bases are shown in the proposed ITS Bases for Section 3.9.

A3 Not used.

A4 CTS 3.8.9 provides the required actions should one or more of the preceding Specifications not be met. CTS 3.8.9 establishes measures that are considered equivalent to the Required Actions of ITS 3.9.1 Condition A, ITS 3.9.2 Condition A; ITS 3.9.2 RA B.1; ITS 3.9.3 RA A.2; ITS 3.9.4 RA A.1, RA A.2 and RA A.3; and ITS 3.9.5 RA A.1, B.1 and B.2. Although the exact wording is not the same, these are considered equivalent actions and adoption of the ITS requirements constitutes an administrative change. In addition, the Completion Time of "immediately" has been annotated on the CTS markup. This is considered implicit in the subject CTS specification. The addition of this immediate Completion Time establishes Required Actions consistent with those specified in the ITS.

A5 CTS Table 4.1-1, Item 6 requires the performance of a channel check of source range instruments at a once per shift frequency. The current Oconee shift duration is 12 hours. ITS SR 3.9.2.1 requires the performance of this channel check at a 12 hour frequency. Therefore this change is administrative in nature and is consistent with the NUREG.

2 A6 Not used.

A7 CTS 3.8.10 states "The reactor building purge system, including the radiation monitor, RIA-45 which initiates purge isolation shall be

tested and verified to be OPERABLE . . . " Although the specification wording specifies testing the system, only the isolation valves provide an accident mitigation function. This is further clarified in the CTS bases for this section which states, "The specification requiring testing of the Reactor Building purge isolation is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products." As a result, CTS 3.8.10 only requires testing the purge supply and exhaust isolation valves. ITS SR 3.9.3.2 requires verifying each Reactor Building Purge supply and exhaust valve actuates to the isolation position on an actual or simulated actuation signal. Therefore this change is considered administrative.

A8 CTS 4.4.4.5 requires verification that the purge isolation valves close in response to a control signal from RIA-45. ITS SR 3.9.3.2 requires verification that the purge isolation valves close on an actual or simulated actuation signal. The requirements for testing the purge
2 isolation functions of RIA-45 is specified in ITS Section 3.3. The combined requirements for testing the purge isolation valves specified in ITS SR 3.9.3.2 and the requirements for testing RIA-45 specified in ITS Section 3.3 are technically equivalent to the testing specified in CTS 4.4.4.5. Therefore this change is an administrative change and is consistent with the NUREG.

A9 Not used.

A10 During fuel loading and refueling operations, CTS 3.8.2 requires one OPERABLE monitor when "core geometry is not being changed," and two OPERABLE monitors "whenever core geometry is being changed." ITS LCO 3.9.2.a requires one source range neutron flux monitor be OPERABLE during the LCO Applicability (MODE 6). ITS LCO 3.9.2.b requires one additional source range neutron flux monitor be OPERABLE during CORE ALTERATIONS and during positive reactivity additions. These changes maintain the requirements of the CTS while providing adequate monitoring capability of changes in the core's neutron flux. This is an administrative change.

ITS Section 3.9

ID 113

Subject: NUREG 3.9-7, SR 3.9.4.1 - changed cross reference to DOC M20 in place of M10.

CTS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

Doc M10

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one DHR loop is in operation and circulating reactor coolant at a flow rate of ≥ [2800] gpm.	12 hours

Supp 2

Doc M20

10

ITS Section 3.9

ID 136

Subject: Revised Applicability statements for LCO 3.9.3, 3.9.4 and 3.9.5 to provide correct indentation and punctuation. Added missing "AND" logical connector after 3.9.5 RA B.2

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by a minimum of four bolts;

b. One door in each air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Reactor Building Purge supply and exhaust isolation signal.

2 APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within
2 containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS AND A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately Immediately

3.9 REFUELING OPERATIONS

3.9.4 Decay Heat Removal (DHR) and Coolant Circulation - High Water Level

LCO 3.9.4 One DHR loop shall be OPERABLE and in operation.

---NOTE---
The required DHR loop may not be in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 21.34 ft above the top of
2 reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. DHR loop requirements not met.	A.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND		
	A.3	Initiate action to satisfy DHR loop requirements.	Immediately
	AND		(continued)

3.9 REFUELING OPERATIONS

3.9.5 Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level

LCO 3.9.5 Two DHR loops shall be OPERABLE, and one DHR loop shall be in operation.

APPLICABILITY: MODE 6 with the water level < 21.34 ft above the top of
2 reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Less than required number of DHR loops OPERABLE.	A.1	Initiate action to restore DHR loop to OPERABLE status.	Immediately
	OR		
	A.2	Initiate action to establish ≥ 21.34 ft of water above the top of reactor vessel flange.	Immediately
B. No DHR loop OPERABLE or in operation.	B.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	B.2	Initiate action to restore one DHR loop to OPERABLE status and to operation.	Immediately
2	AND		(continued)

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Specification 3.10.1

A1 except as marked

Suppl. 2

TABLE 4.20-1
SSF INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

		Check	Calibrate	Remarks
1.	RCS Pressure (3)	~~WE~~ 7 days	18 months	Loop ~~A, B~~
2.	SSF RC Makeup Pump (3)			
	Suction Pressure	QU(1)	18 months	
	Discharge Pressure	QU(1)	18 months	
	Suction Temperature	QU(1)	18 months	
	Discharge Flow	QU(1)	18 months	
3.	RC System Temperature (3)	~~NA(2)~~	18 months	Loop ~~A, B~~ Hot, Cold
4.	Pressurizer Water Level (3)	~~WE~~ 7 days	18 months	
5.	SSF Auxiliary Service Water Pump			
	Suction Pressure	QU(1)	AN	
	Discharge Pressure	QU(1)	AN	
	Unit 1 Discharge Pressure	NA	AN	
	Unit 2 Discharge Pressure	NA	AN	
	Unit 3 Discharge Pressure	NA	AN	
	Discharge Test Flow	QU(1)	AN	
	Suction Temperature	QU(1)	AN	
6.	Steam Generator Levels (3)	~~WE~~ 7 days	18 months	A, B
7.	Underground Fuel Oil Storage Tank Inventory	NA	AN	
8.	D/G Service Water Pump			
	Discharge Flow	QU(1)	AN	
	Discharge Pressure	QU(1)	AN	
9.	D/G Air Start System Pressure	WE	AN	

(1) Check when pump operated/tested per IST.

(2) This instrumentation is normally aligned through a transfer/isolation device to each Unit Control Room and is thus checked in accordance with Specification 4.1, Table 4.1-1, Item 7. Every 18 months, the instrument string to the SSF Control Room will be checked and calibrated.

(3) Units 1, 2, 3.

9 of 9

Amendment No. ~~228~~ (Unit 1)
Amendment No. 229 (Unit 2)
Amendment No. 225 (Unit 3)

ITS Section 3.6 3.7 4.0 5.0

ID 139

Subject: Revised ITS to remove Penetration Room Ventilation System Tech Spec change 95-04

Specification 4.0

LA2

5.2 CONTAINMENT

Specification

Suppl. 2

The containment for this unit consists of three systems which are the reactor building, reactor building isolation system, and penetration room ventilation system.

5.2.1 Reactor Building

The reactor building completely encloses the reactor and its associated reactor coolant system. It is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and flat foundation slab. The cylindrical portion is prestressed by a post tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post tensioning system. The structure can withstand the loss of 3 horizontal and 3 vertical tendons in the cylinder wall or adjacent tendons in the dome without loss of function. The foundation slab is conventionally reinforced with high strength reinforcing steel. The entire structure is lined with 1/4" welded steel plate to provide vapor tightness.

The internal volume of the reactor building is approximately 1.836 x 10^6 cu. ft. The approximate inside dimensions are: diameter--116'; height--208 1/2'. The approximate thickness of the concrete forming the building are: cylindrical wall--3 3/4'; dome--3 1/4'; and the foundation slab--8 1/2'.

The concrete containment structure provides adequate biological shielding for both normal operation and accident situations. Design pressure and temperature are 59 psig and 286°F, respectively.

The reactor building is designed for an external atmospheric pressure of 3.0 psi greater than the internal pressure. This is greater than the differential pressure of 2.5 psig that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. Since the building is designed for this pressure differential, vacuum breakers are not required.

Penetration assemblies are seal welded to the reactor building liner. Access openings, electrical penetrations, and fuel transfer tube covers are equipped with double seals. Reactor building purge penetrations and reactor building atmosphere sampling penetrations are equipped with double valves having resilient seating surfaces. (1)

4 of 7

Amendment No. 189 (Unit 1)
Amendment No. 189 (Unit 2)
Amendment No. 186 (Unit 3)

Specification 4.0

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in FSAR Section 15 with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks.

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. (2)

5.2.3 Penetration Room Ventilation System

This system is designed to collect, control, and minimize the release of radioactive materials from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slight negative pressure will be maintained in the penetration room to assure inleakage. (3)

REFERENCES

(1) FSAR Sections 6.2.1, and 6.2.3

(2) FSAR Section 6.2.3

(3) FSAR Section 6.5.1.1

LA 2

ITS Section 3.0 3.1 3.2 3.3 3.4 3.5 3.6 3.7
3.9 3.10 5.0

ID 13

Subject: Revised ITS to incorporate TSC 98-02 dated 2/2/98 that changed refueling outage frequency to 18 month frequency. Also incorporates latest revision to TSC 95-03, Penetration Room Ventilation System.

Specification 5.0

(A1 <except as marked>)

Add 5.5.9 — (A37)

4 SURVEILLANCE REQUIREMENTS

4.0 SURVEILLANCE STANDARDS

Applicability

Applies to surveillance requirements which relate to tests, calibrations and inspections necessary to assure that the quality of structures, systems and components is maintained and that operation is within the safety limits and limiting conditions for operation.

Objective

To specify minimum acceptable surveillance requirements.

<SEE 3.0>

Specification

4.0.1 Surveillance of structures, systems, components and parameters shall be as specified in the various subsections to this Technical Specification section, Section 4.0, except as permitted by Technical Specifications 4.0.2 and 4.0.3 below.

4.0.2 Minimum surveillance frequencies, unless specified otherwise, may be adjusted as follows to facilitate test scheduling:

Specified Frequency	Maximum Allowable Interval Between Surveillances
Five times per week	2 days
Two times per week	5 days
Weekly	10 days
Bi-Weekly	20 days
Monthly	45 days
Bi-Monthly	90 days
Quarterly	135 days
Semiannually	270 days
Annually	18 months
18 months	22 months, 15 days
Refueling Outage	22 months, 15 days

Supp 2

Clarifying words in individual specifications such as "every," "at least," or "at least once every" are not intended to alter the frequencies defined by this specification.

4.0.3 If conditions exist such that surveillance of an item is not necessary to assure that operation is within the safety limits and limiting conditions for operation, surveillance need not be performed if such conditions continue for a length of time greater than the specified surveillance interval. Surveillance waived as a result of this specification shall be performed prior to returning to conditions for which the surveillance is necessary to assure that operation is within safety limits and limiting conditions for operation.

4.0.4 Inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

Specification 5.0

(A1) (except as marked)

4.5.4 Penetration Room Ventilation System

Applicability

Applies to testing of the Penetration Room Ventilation System

Objective

To verify that the Penetration Room Ventilation System is operable.

Specification (see 3.7.)

4.5.4.1 Operational and Performance Testing

a. Monthly, each train of the Penetration Room Ventilation System shall be operated for at least 15 minutes at design flow ±10%.

Supp. 2

b. Every 18 months, it shall be demonstrated that:

Supp. 1

5.5.12.a1. The Penetration Room Ventilation System fans operate at design flow (± 10%) when tested in accordance with ANSI N510-1975.

5.5.12.f 2. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than six inches of water at the system design flow rate (± 10%).

3. Each branch of the Penetration Room Ventilation System is capable of automatic initiation.

4. The bypass valve for filter cooling is manually operable. (see 3.7)

c. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the Penetration Room purge filters:

Supp. 2

1. Every 18 months;

2. After each complete or partial replacement of a HEPA filter bank or charcoal adsorber bank; (LA21)

3. After any structural maintenance on the system housing;

4. After painting, fire, or chemical release in any ventilation zone communicating with the system.

5.5.12.a 5.5.12.c d. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

The provisions of SR 3.0.2 & SR 3.0.3 are applicable. (A28) (MS)

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Specification 5.0

(CA21)

Supp 2

c. Every 18 months, or following 720 hours of system operation, or after painting, fire, or chemical release in any ventilation zone communicating with the system, a carbon sample shall be removed from the Penetration Room Ventilation system filters for laboratory analysis. Within 31 days of removal this sample shall be verified to show ≥90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable. (See 3.7)

5.5.12.c

Supp 2

Bases

Pressure drop across the combined high efficiency particulate air (HEPA) filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per year operating cycle establishes performance capability.

(HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system every month will demonstrate operability of the filters and adsorber system. Operation for 15 minutes demonstrates operability and minimizes the moisture build up during testing.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

(A2)

30 of 68

Specification 5.0

(A1 except as marked)

4.5.5 Low Pressure Injection System Leakage

(A1 except as marked)

Applicability

Applies to Low Pressure Injection System leakage.

Objective

To maintain a preventive leakage rate for the Low Pressure Injection System which will prevent significant off-site exposures.

Specification

(CA18)

4.5.5.1 Acceptance Limit

The maximum allowable leakage from the Low Pressure Injection System components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

Spec 2 1

4.5.5.2 Test

Every 18 months, the following tests of the Low Pressure Injection System shall be conducted to determine leakage:

a. The portion of the Low Pressure Injection System, except as specified in (b), that is outside the containment shall be tested either by use in normal operation or by hydrostatically testing at 350 psig.

b. Piping from the containment emergency sump to the low pressure injection pump suction isolation valve shall be pressure tested at no less than 59 psig.

c. Visual inspection shall be made for excessive leakage from components of the system. Any excessive leakage shall be measured by collection and weighing or by another equivalent method.

Bases

The leakage rate limit for the Low Pressure Injection System is a judgement value based on assuring that the components can be expected to operate with-out mechanical failure for a period on the order of 200 days after a loss of coolant accident. The test pressure (350 psig) achieved either by normal system operation or by hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the pressure test for the return lines from the containment to the Low Pressure Injection System (59 psig) is equivalent to the design pressure of the containment. The dose to the thyroid calculated as a result of this leakage is 0.76 rem for a two-hour exposure at the site boundary.

(A2)

REFERENCE

FSAR, Section 15.15.4, and 6.3.3.2.2

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ITS Section 5.0

ID 85

Subject: Revised ITS 5.2.2.c and d regarding the time period allowed to fill vacant position.

5.2 Organization (continued)

5.2.2 Station Staff

a. A non-licensed operator shall be onsite for each reactor containing fuel and an additional non-licensed operator shall be onsite for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

b. At least one licensed Reactor Operator (RO) per unit shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.g
2 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

2 d. A Radiation Protection Technician shall be on site when fuel
2 is in the reactor. The position may be vacant for not more
2 than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, Non-Licensed operators, and key maintenance personnel).

Any deviation from the above guidelines shall be authorized in advance by the Station Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be periodically reviewed to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The Operations Superintendent or Shift Operations Manager shall hold an SRO license.

(continued)

5.2 Organization

5.2.2 Station Staff (continued)

g. The Shift Work Manager, whose functions include those of a Shift Technical Advisor (STA), shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the Shift Work Manager shall meet the qualifications for STA specified by the Commission Policy Statement on Engineering Expertise on Shift.

h. The qualified manpower necessary for achieving alternate shutdown using the Standby Shutdown Facility (SSF) will be available at the plant at all times. The manpower necessary to operate the SSF will be exclusive of the fire brigade and the minimum operating shift that is required to be present in the Control Room.

Specification 5.0

(A1) <except as marked>

ADDITIONAL REQUIREMENTS

5.2.2.b 1. One licensed operator per unit shall be in the Control Room at all times when there is fuel in the reactor vessel. 10CFR50.54(m) A15

2. Two licensed operators shall be in the Control Room during startup and scheduled shutdown of a reactor. A15

3. At least one licensed operator shall in the reactor building when fuel handling operations in the reactor building are in progress. LA4

4. An operator holding a Senior Reactor Operator license and assigned no other operational duties shall be in direct charge of refueling operations. A15

Supp 2 | 5.2.2.d 5. At least one Radiation Protection Technician per shift shall have sufficient training to perform routine health physics requirements. 10CFR55.59(m) M19

6. If the computer for a reactor is inoperable for more than eight hours, an operator, in addition to those required in (1) and (2) above, shall supplement the Control Room shift crew. LA4

5.2.2.b 7. An operator holding a Senior Reactor Operator's license shall be in the Control Room from which the unit is operated whenever the unit is above cold shutdown. M17

Supp 2 | 5.2.2.c 8. Temporary deviations for a period of time not to exceed 2 hours from the requirements of Table 6.6-1 may be allowed in cases of sudden illness, injury or other similar emergencies provided replacement personnel are notified immediately and are on site as soon as possible to return shift manning to minimum. unexpected absence; immediate action is taken
5.2.2.d

5.2.2.h 9. The qualified manpower necessary for achieving alternate shutdown using the SSF will be available at the plant at all times. The manpower necessary to operate the SSF will be exclusive of the fire brigade and the 3 member minimum operating shift that is required to be present in the Control Room.

M14 With an inoperable channel for the specified function inoperable for > 7 days, CTS Table 3.5.6-1 Action 3 requires submittal of a report to the commission within the next 30 days. ITS 5.6.6 requires submittal of the report within the next 14 days and also requires outlining the preplanned alternative method of monitoring. The requirements to submit the report within 14 days and outline the preplanned alternative method of monitoring are more restrictive requirement upon unit operation and are consistent with the NUREG. This change is acceptable since the additional requirements are reasonable and are not a significant impact on unit operations.

M15 CTS 6.9.2 includes a listing of NRC approved analytical methodologies which are used to determine ITS values specified in the COLR but does not include DPC-NE-3005-P, UFSAR Chapter 15 Transient Analysis Methodology, DPC, JUL97. ITS 5.6.5.b includes this reference since it is the methodology which is used to determine DNBR parameters specified as being in the COLR in ITS 3.4.1. The methodology has been submitted separately for NRC approval. Although listed in ITS 5.6.5.b, this methodology has not been approved by the NRC at the time of the ITS submittal. The addition of this methodology in ITS 5.6.5.b is a more restrictive requirement upon unit operation and is consistent with the NUREG. This is acceptable since the specified methodology reflects the methodology to be utilized for determining the associated values.

M16 A CTS provision comparable to ITS 5.5.1 requirements regarding licensee initiated changes to the ODCM does not exist. ITS 5.5.1 specifies requirements regarding documentation and record keeping for review of changes to the ODCM, specifies approval authority for changes to the ODCM and specifies reporting requirements for changes to the ODCM. These additional requirements are a more restrictive requirement upon unit operation and are consistent with the NUREG. The additional requirements are acceptable since they are reasonable and are not a significant administrative burden.

2 M17 CTS Table 6.1-1, Additional Requirement 8 allows temporary deviations
2 from shift manning requirements in cases of sudden illness, injury or
2 other similar emergencies provided replacement personnel are notified
2 immediately and are on site as soon as possible to return shift manning
2 to minimum. ITS 5.2.2.c and 5.2.2.d provides a similar requirement but
2 specifies the position(s) be filled within two hours. The ITS is
2 considered more restrictive since a specific time period has been
2 specified that could be less time than is possible to fill the vacant
2 position, depending upon the conditions. The requirement is acceptable
2 since it is reasonable and is not a significant burden on plant
2 operations. The change is consistent with the NUREG.

5.2 Organization

5.2.2 Unit Staff (continued)

shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. A [Health Physics Technician] shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

(continued)

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes are made for clarity or for consistency with the Improved Technical Specifications (ITS) Writer's Guide.

3 The requirement/statement are deleted since it is not applicable to this facility. The following requirements are renumbered, where applicable, to reflect this deletion.

4 Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5 NUREG Specification 5.1.2 is not adopted in the ITS, consistent with the current licensing basis. The control room command function is adequately addressed in 10 CFR 50.54(m)(2)(ii).

6 The position of Senior Vice-President, Nuclear Generation Department is added as ITS 5.2.1.d. This change is consistent with the current licensing basis.

7 The NUREG 5.2.2 introductory phrase "The unit staff organization shall include the following:" is omitted. This phrase provides no requirements or clarification, and implies that "the following" is intended to be a listing of required organizational elements. However, also included are general requirements for the staff, e.g., absence and overtime limitations, etc. Therefore, the introductory phrase is not appropriate.

8 The NUREG 5.2.2.a term "*assigned to* each reactor containing fuel" is replaced with the more prescriptive terminology "*on site* for each reactor containing fuel." This provides clarification that the individual must be physically present on site in lieu of "being assigned." The words "per unit" is added to 5.2.2.b to clarify the an RO is required to be present for each unit with fuel in the reactor.

2 9 Not used.

2 10 Not used.

ITS Section 5.0

ID 86

Subject: Revised ITS 5.2.2.d to use terminology "Radiation Protection Technician."

5.2 Organization (continued)

5.2.2 Station Staff

a. A non-licensed operator shall be onsite for each reactor containing fuel and an additional non-licensed operator shall be onsite for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

b. At least one licensed Reactor Operator (RO) per unit shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. 2

d. A Radiation Protection Technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position. 2 2 2

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, Non-Licensed operators, and key maintenance personnel).

Any deviation from the above guidelines shall be authorized in advance by the Station Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be periodically reviewed to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

f. The Operations Superintendent or Shift Operations Manager shall hold an SRO license.

(continued)

Specification 5.0

(A1) <except as marked>

ADDITIONAL REQUIREMENTS

5.2.2.b 1. One licensed operator per unit shall be in the Control Room at all times when there is fuel in the reactor vessel. 10CFR50.54(m) A15

2. Two licensed operators shall be in the Control Room during startup and scheduled shutdown of a reactor. A15

3. At least one licensed operator shall in the reactor building when fuel handling operations in the reactor building are in progress. LA4

4. An operator holding a Senior Reactor Operator license and assigned no other operational duties shall be in direct charge of refueling operations. A15

Radiation Protection Technician 10CFR55.59(m) M19

Supp 2 | 5.2.2.d 5. At least one person per shift shall have sufficient training to perform routine health physics requirements.

6. If the computer for a reactor is inoperable for more than eight hours, an operator, in addition to those required in (1) and (2) above, shall supplement the Control Room shift crew. LA4

5.2.2.b 7. An operator holding a Senior Reactor Operator's license shall be in the Control Room from which the unit is operated whenever the unit is above cold shutdown. M17

Supp 2 | 5.2.2.c 8. Temporary deviations from the requirements of Table 6.6-1 may be allowed in cases of sudden illness, injury or other similar emergencies, provided replacement personnel are notified immediately and are on site as soon as possible to return shift manning to minimum. for a period of time not to exceed 2 hours; unexpected absence; immediate action is taken

5.2.2.d

5.2.2.h 9. The qualified manpower necessary for achieving alternate shutdown using the SSF will be available at the plant at all times. The manpower necessary to operate the SSF will be exclusive of the fire brigade and the 3 member minimum operating shift that is required to be present in the Control Room.

2 M18 CTS 6.1.1.4 states that minimum qualifications for station staff are encompassed by ANSI/ANS-3.1-1978 except the Radiation Protection Manager, the Operations Superintendent and the Shift Operations Manager. The second paragraph of CTS 6.1.1.4 specifies the minimum qualifications for the Radiation Protection Manager and the third paragraph allows the station manager to appoint someone of lesser qualifications provided it is based on recommendations from the Nuclear Technical Services Manager and approved by the Site Vice President. ITS 5.3.1 specifies that the Radiation Protection Manager meet the minimum qualifications of ANSI/ANS-3.1-1978. No exceptions are provided. The requirement is acceptable since it is reasonable and is not a significant burden on plant operations. The change is consistent with the NUREG.

2 M19 CTS Table 6.1-1, Additional Requirement 5 specifies that at least one person per shift shall have sufficient training to perform routine health physics requirements. ITS 5.2.2.d specifies that a Radiation Protection Technician shall be on site. Since the ITS only allows a Radiation Protection Technician to fulfill the previous requirement for the person on shift where before any individual with sufficient training to perform routine health physics requirements was adequate, the change is more restrictive. This more restrictive change is an acceptable restriction on plant operation and is consistent with the NUREG.

2 M20 License Condition H of Oconee Operating Licenses for Units 1, 2, and 3 requires the licensee to implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. ITS 5.5.3 provides comparable requirements and specifies the systems included in the program. Even though these systems are currently identified in the program that implements this requirement, specifying the systems is more restrictive since these systems were not previously identified by technical specifications as being required to be included. The change is consistent with the NUREG.

TECHNICAL CHANGES - LESS RESTRICTIVE

L1 CTS 3.6.7.1.c requires submittal of a report for abnormal degradation of containment tendons within 15 days. ITS 5.6.7 requires a Tendon Surveillance Report within 30 days. The additional time to submit the report is a less restrictive requirement upon plant operation and is consistent with the NUREG. This change is acceptable, however, because there is no requirement for the NRC to approve these reports. Completion and submittal of the reports is clearly not necessary to ensure safe operation of the unit during the additional time intervals provided by this change. This change is consistent with the NUREG.

L2 CTS Specifications 6.6.1.3 (Personnel Exposure and Monitoring Report) and 6.6.1.5 (Annual Radiological Environmental Operating Report), which require that these reports be submitted by March 1 and May 1 of each year, respectively, are revised in ITS Specifications 5.6.1 and 5.6.2 to permit these reports to be submitted by April 30 and May 15 of each year, respectively. This is a relaxation of requirements, which is less restrictive. This change is acceptable, however, because these reports cover the previous calendar year, and there is no requirement for the NRC to approve these reports. Completion and submittal of the reports is clearly not necessary to ensure safe operation of the unit during the additional time intervals provided by these changes. This change is consistent with the NUREG.

L3 CTS Specification 6.1.1.4, which requires that the Operations Superintendent hold, or have held, an SRO license, and, that the Shift Operations Manager hold an SRO license, is revised in ITS Specification 5.2.2.f to require that either the Operations Superintendent" or the Shift Operations Manager hold an SRO license. The Shift Operations Manager is an off-shift manager reporting to the Operations Superintendent who can be delegated the Operations Superintendent duties. This is a relaxation of requirements, which is less restrictive. 10 CFR 55 requires that an individual that directs the licensed activities of licensed operators hold an SRO license. This change is acceptable, however, since the Technical Specification requirements will continue to assure that an individual with direct control of the operations shift crews holds an SRO license. This change is consistent with the NUREG.

L4 CTS 4.4.1.2.3 requires leakage from penetrations and isolation valves (Type B and C tests) to be $\leq$ 0.10 weight percent of containment post-accident air mass per 24 hours (equivalent to 0.50 La). This limit is applicable whenever containment integrity is required. ITS 5.5.2, Containment Leakage Rate Testing Program, specifies the acceptance criterion for Type B and C testing as $\leq$ 0.60 La during the first unit

startup following testing in accordance with this program. At other times the acceptance criterion for Type B and C testing is ≤ 1.0 La. The increase in the acceptance criterion from 0.50 La to 0.60 La during the first unit startup following leak rate testing and from 0.50 La to 1.0 La at other times are less restrictive requirements upon unit operation and are consistent with the NUREG. Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the combined Type B and C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a. Compliance with these requirements ensure a containment configuration, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

L5 A CTS provision to apply SR 3.0.2, the SR interval extension, to the Tendon Surveillance Testing does not exist. Although CTS 4.0.2 provides surveillance extensions comparable to SR 3.0.2, there is no listed maximum allowable interval for the specified 5 year testing frequency. Hence, the CTS does not permit an "extension" to the specified tendon testing frequency. ITS 5.5.7 permits applying the SR 3.0.2 Frequency extension to the inspections required by the Containment Tendon Testing Program. Applying the SR 3.0.2 Frequency extension to the inspections required by the Containment Tendon Testing Program is a less restrictive requirement upon unit operation and is consistent with the NUREG.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities). The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs.

5.2 Organization

5.2.2 Unit Staff (continued)

shall be assigned (onsite) (8) for each control room from which a reactor is operating in MODES 1, 2, 3, or 4. (1)

[Two unit sites with both units shutdown or defueled require a total of three non-licensed operators for the two units.] (per unit) (8)

b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room. T6.1-1 Add Req 1 T6.1-1 Add Req 7

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements. T6.1-1 Add Req 8

(1) Supp 2 — Radiation Protection Technician

d. A [Health Physics Technician] shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position. T6.1-1 Add. Req 5 & 8

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, health physicists, auxiliary operators, and key maintenance personnel). (NON-Licensed)

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an [8 or 12] hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed: (11) G.M.4.b

(continued)

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes are made for clarity or for consistency with the Improved Technical Specifications (ITS) Writer's Guide.

3 The requirement/statement are deleted since it is not applicable to this facility. The following requirements are renumbered, where applicable, to reflect this deletion.

4 Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5 NUREG Specification 5.1.2 is not adopted in the ITS, consistent with the current licensing basis. The control room command function is adequately addressed in 10 CFR 50.54(m)(2)(ii).

6 The position of Senior Vice-President, Nuclear Generation Department is added as ITS 5.2.1.d. This change is consistent with the current licensing basis.

7 The NUREG 5.2.2 introductory phrase "The unit staff organization shall include the following:" is omitted. This phrase provides no requirements or clarification, and implies that "the following" is intended to be a listing of required organizational elements. However, also included are general requirements for the staff, e.g., absence and overtime limitations, etc. Therefore, the introductory phrase is not appropriate.

8 The NUREG 5.2.2.a term "*assigned to* each reactor containing fuel" is replaced with the more prescriptive terminology "*on site* for each reactor containing fuel." This provides clarification that the individual must be physically present on site in lieu of "being assigned." The words "per unit" is added to 5.2.2.b to clarify the an RO is required to be present for each unit with fuel in the reactor.

2 9 Not used.

2 10 Not used.

ITS Section 5.0

ID 87

Subject: Revised 5.3.1 to incorporate NRC comments regarding minimium qualifications for plant staff. Remove exceptions for Rad Prot Mgr.

5.0 ADMINISTRATIVE CONTROLS

5.3 Station Staff Qualifications

5.3.1 Each member of the station staff shall meet or exceed the minimum
2 qualifications described in Section 4 of ANSI/ANS-3.1-1978,
"Selection and Training of Nuclear Power Plant Personnel" except
2 for the Operations Superintendent and the Shift Operations Manager.

The Operations Superintendent shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training, on a one-for-one time basis.

The Shift Operations Manager shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training on a one-for-one time basis.

Specification 5.0

(A1) <except as marked>

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, REVIEW, AND AUDIT

6.1.1 Organization

5.2.1 Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offset organizations shall include the positions for activities affecting the safety of the nuclear power plant.

6.1.1.1 Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. 5.2.1.a These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR.

5.1.1 6.1.1.2 The Station Manager shall be responsible for overall plant safe operation and shall have control over 5.2.1.b those onsite activities necessary for safe operation and maintenance of the plant and shall delegate in writing the succession to this responsibility during his absence.

6.1.1.3 The Vice President, Oconee Nuclear Site, shall have corporate responsibility for overall plant nuclear 5.2.1.c safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

5.2.1.d The Senior Vice President, Nuclear Generation Department will be the Senior Nuclear Executive and have corporate responsibility for overall nuclear safety.

5.3.1 6.1.1.4 Incorporated in the staff of the station shall be personnel meeting the minimum requirements encompassing the training and experience described in Section 4 of ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel" except for the Radiation Protection Manager, the Operations Superintendent and the Shift Operations Manager. (M18)

The Radiation Protection Manager shall have a bachelor's degree in a science or engineering subject or the equivalent in experience, including some formal training in radiation protection, and shall have at least five years of professional experience in applied radiation protection of which three years shall be in applied radiation protection work in one of Duke Energy Corporation's nuclear stations.

A qualified individual who does not meet the above requirements, but who has demonstrated the required radiation protection management capabilities and professional experience in applied radiation protection work at one of Duke Energy Corporation's multi-unit nuclear stations, may be appointed to the position of Radiation Protection Manager by the Station Manager, based on the recommendations of the Nuclear Technical Services Manager and as approved by the Site Vice President, Nuclear Operations.

5.3.1 The Operations Superintendent shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training, on a one-for-one time basis. The Operations Superintendent 5.2.2.f shall hold or have held a Senior Reactor Operator license. (L3)

Add 5.5.15 (M1)

Add 5.5.16

Amendment No. 226 (Unit 1)
Amendment No. 226 (Unit 2)
Amendment No. 223 (Unit 3)

Specification 5.0

(A1) <except as marked>

5.3.1 The Shift Operations Manager shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training on a one-for-one time basis. The Shift Operations Manager shall hold a Senior Reactor Operator license.

5.2.2.f

Operations Superintendent or (L3)

6.1.1.5 The individuals who train the operating staff and those who carry out radiation protection and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.1.e

6.1.1.6 Minimum operating shift crew requirements shall be as specified in Table 6.1-1. (A1)

(LA5)

6.1.1.7 Retraining and replacement of station personnel shall be in accordance with Section 5.5 of the ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."

6.1.1.8 A training program for the fire brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions may be held quarterly.

(LA17)

6.1.1.9 The Shift Manager is an experienced SRO, who has been instructed in additional academic subjects, and will be assigned on-shift to provide the accident assessment capability.

5.2.2.g

Supp 1

The operating experience assessment function will be provided by the Station Safety Review Group. (A5)

INSERT 5.2.2.g from ITS (M10)

implementation is covered by a more generic item, ITS 5.4.1.e, which requires this activity for Programs specified in ITS 5.5 (including the ODCM). Therefore, it is not necessary to specifically identify each program. Since the requirements remain in Technical Specifications, this is considered to be a change in the method of presentation only and, therefore, is considered an administrative change. This change is consistent with the NUREG.

A25 CTS 6.2.3 requirements associated with reporting requirements are not retained in the ITS since they duplicates requirements contained in the regulations. Such duplication is unnecessary and results in additional administrative burden to revise the duplicate TS when these regulations are revised. Since removal of the duplication results in no actual change in the requirements, removal of the duplicative information is considered an administrative change. Further, changes to the requirements are controlled by the NRC. This change is consistent with the NUREG.

A26 CTS 6.4.4.b uses the term "nuclear equipment operator" in a list of examples regarding the scope of requirements regarding limiting the working hours of specified plant personnel. ITS 5.2.2.e refers to this position as "non-licensed operators" and reflects current plant terminology as well as provide consistency with other references within the NUREG and regulations. Since the change in the terminology does not result in an actual or interpretational change, this change is administrative in nature.

2 A27 Not used.

A28 CTS 4.0.2 permits application of a frequency extension for specified frequencies. The frequencies for testing the CRVS Booster Fan Train filters and Penetration Room Ventilation System filters are addressed by CTS 4.0.2. Accordingly, CTS permits frequency extensions for the CRVS Booster Fan Train filters and Penetration Room Ventilation System filters. ITS SR 3.0.2 permits extending SR Frequencies. This change is an administrative change and is consistent with the NUREG.

A29 CTS 4.0.4 requires testing of ASME Code Class 1, 2 and 3 pumps and valves in accordance with Section XI of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction. This requirement is not retained in ITS since it duplicates applicable regulations. Since the requirements are directly

tied to the requirements of 10 CFR 50.55a, elimination of CTS 4.0.4 is a less restrictive change and is consistent with the NUREG.

A30 An explicit CTS requirement for a program to monitor containment tendon degradation does not exist. CTS 4.4.2 specifies details associated with containment tendon testing. The program required in ITS 5.5.7 meets the intent of CTS 4.4.2. The details of CTS 4.4.2 are relocated to the Pre-stressed Concrete Containment Tendon Surveillance Program. The requirement to establish a program to accomplish existing requirements is an administrative requirement and is consistent with the NUREG.

A31 CTS 4.0.2 permits application of a frequency extension for specified frequencies. The frequencies for determining the quantity of radioactive material in the Liquid Holdup Tanks and Waste Gas Holdup Tanks are addressed by CTS 4.0.2. Accordingly, CTS permits frequency extensions for the determination of the quantity of radioactive material in the Liquid Holdup Tanks and Waste Gas Holdup Tanks. An explicit CTS requirement for the surveillance of the hydrogen concentration in the Waste Gas Holdup Tanks does not exist. ITS SR 3.0.2 permits extending SR Frequencies. Therefore, this change is an administrative change and is consistent with the NUREG.

A32 An explicit CTS requirement for a program to monitor explosive gas and radioactivity levels in storage tanks does not exist. CTS 3.9 and 3.10 specify details associated with explosive gas and radioactivity levels in storage tanks. ITS 5.5.13 requires establishment of a program which provides controls for explosive gas and storage tank radioactivity monitoring. The program required in ITS 5.5.13 meets the intent of CTS 3.9 and 3.10. The details of CTS 3.9 and 3.10 are relocated to the UFSAR. The requirement to establish a program to accomplish existing requirements is an administrative requirement and is consistent with the NUREG.

A33 CTS 3.9.2 and CTS 3.10.3 are not retained. CTS 3.9.2 and CTS 3.10.3 state that CTS 3.0 does not apply. CTS 3.0 requires the unit be shutdown if the LCO or associated Action statement cannot be met due to circumstances in excess of those addressed in the specification. The requirements of CTS 3.0 are comparable to ITS LCO 3.0.3. ITS LCO 3.0.3 is not applicable to the programs in ITS section 5.5 unless specifically provided for in the program description in ITS 5.5. The non-retention of CTS 3.9.2 and 3.10.3 is an administrative change and is consistent with the NUREG.

2 A34 Not used.

A35 Limits for Specifications 3.1.1, Shutdown Margin; 3.1.3 Moderator Temperature Coefficient (MTC); and 3.9.1, Boron Concentration are added to the listing of ITS limits specified in the COLR. The relocations of these limits to the COLR are addressed in the conversion packages for these Specifications. The addition of these Specifications to the listing in ITS 5.6.5 is an administrative change and is consistent with the NUREG.

A36 CTS 4.1.4 requires radioactive effluent monitoring instrumentation to be tested as stated in FSAR Chapter 16. Since there are no Surveillance Requirements located in CTS, this serves as only a pointer to the FSAR Chapter 16 requirement and is deleted. Requirements for radiological effluent instrumentation were previously relocated from the CTS to the UFSAR by 187/187/184 on 1/22/91. This deletion is considered administrative and is consistent with the NUREG.

A37 An explicit CTS provision regarding an Inservice Testing Program for inservice testing of ASME Class 1, 2 and 3 pumps and valves does not exist. ITS 5.5.9 specifies provisions regarding the Inservice Testing Program for inservice testing of ASME Class 1, 2 and 3 pumps and valves. The requirements for testing ASME Class 1, 2 and 3 pumps and valves is separately required by 10 CFR 50.55a. Therefore, the inclusion of the ITS 5.5.9 provisions for the Inservice Testing Program is an administrative change and is consistent with the NUREG.

A38 An explicit CTS requirement for a program to implement required testing of diesel fuel does not exist. CTS 5.5.14 requires a establishment of a program to implement required testing of diesel fuel. The requirement to establish a program to accomplish existing requirements is an administrative requirement and is consistent with the NUREG.

A39 CTS 6.4.4.e.6 refers to limitations on the operability of liquid and gaseous treatment systems. ITS 5.5.5.f refers to functional capability in lieu of operability. ITS generally limits the term OPERABILITY to use within ITS LCOs, since it is an ITS defined term. This is a change in terminology only. No change in application or interpretation is intended. This is an administrative change and is consistent with the NUREG.

A40 CTS 6.4.4.e.6 refers to annual dose or dose commitments contained in Appendix I to 10 CFR Part 50 as clarified in FSAR Chapter 16. The phrase as clarified in FSAR Chapter 16 is not retained since it is unnecessary. The program being described in this section is required to be contained in UFSAR Chapter 16. This is an administrative change and is consistent with the NUREG.

A41 CTS 3.10.1 requires limiting the radioactive quantity in each waste gas holdup tank to ≤ 3.8E+05 Curies noble gas (considered as Xe-133). ITS 5.5.13.b requires the quantity in each gas storage tank to be less than the amount which would result in a whole body exposure of ≥ 0.5 rem to an individual at the nearest exclusion area boundary in the event of an uncontrolled release of the tank's contents. The CTS specified limit on tank contents represents a quantity which results on the ITS limit on whole body exposure. This is an administrative change and is consistent with the NUREG.

2 A42 CTS 6.4.1.d requires that written procedures with appropriate check-off
2 lists and instructions shall be provided for "emergency procedures
2 involving potential or actual release of radioactivity." ITS 5.4.1
2 provides comparable requirements with the exception of referring to
2 emergency operating procedures required to implement the requirements of
2 NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-
2 33. Since these requirements are equivalent, this is considered a
2 change in presentation only and is therefore administrative. The change
2 is consistent with the NUREG.

2 M18 CTS 6.1.1.4 states that minimum qualifications for station staff are encompassed by ANSI/ANS-3.1-1978 except the Radiation Protection Manager, the Operations Superintendent and the Shift Operations Manager. The second paragraph of CTS 6.1.1.4 specifies the minimum qualifications for the Radiation Protection Manager and the third paragraph allows the station manager to appoint someone of lesser qualifications provided it is based on recommendations from the Nuclear Technical Services Manager and approved by the Site Vice President. ITS 5.3.1 specifies that the Radiation Protection Manager meet the minimum qualifications of ANSI/ANS-3.1-1978. No exceptions are provided. The requirement is acceptable since it is reasonable and is not a significant burden on plant operations. The change is consistent with the NUREG.

2 M19 CTS Table 6.1-1, Additional Requirement 5 specifies that at least one person per shift shall have sufficient training to perform routine health physics requirements. ITS 5.2.2.d specifies that a Radiation Protection Technician shall be on site. Since the ITS only allows a Radiation Protection Technician to fulfill the previous requirement for the person on shift where before any individual with sufficient training to perform routine health physics requirements was adequate, the change is more restrictive. This more restrictive change is an acceptable restriction on plant operation and is consistent with the NUREG.

2 M20 License Condition H of Oconee Operating Licenses for Units 1, 2, and 3 requires the licensee to implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. ITS 5.5.3 provides comparable requirements and specifies the systems included in the program. Even though these systems are currently identified in the program that implements this requirement, specifying the systems is more restrictive since these systems were not previously identified by technical specifications as being required to be included. The change is consistent with the NUREG.

INSERT 5.0-5A

Each member of the station staff shall meet or exceed the minimum
2 qualifications described in Section 4 of ANSI/ANS-3.1-1978, "Selection
2 and Training of Nuclear Power Plant Personnel" except for the Operations Superintendent and the Shift Operations Manager.

The Operations Superintendent shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training, on a one-for-one time basis.

The Shift Operations Manager shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training on a one-for-one time basis.

11 NUREG 5.2.2.e is modified to retain the current licensing basis regarding controls to limit working hours. Additionally, the term "auxiliary operator" is replaced with "non-licensed operator." This change is editorial to reflect current plant terminology.

12 Plant specific terminology is utilized in ITS 5.2.2.g for clarity. The shift work manager provides the accident assessment function. This is consistent with the current licensing basis.

2 13 NUREG 5.3.1 is modified to retain exceptions to qualification
2 requirements for the Operations Superintendent and the Shift Operations
2 Manager.

2 14 Not used.

15 The Radiological Effluent Technical Specifications were previously relocated to Chapter 16 of the UFSAR, rather than the ODCM. Section 5.5.1.b of the NUREG which addresses these controls has been included in ITS 5.5.5. ITS 5.5.5 includes controls for chapter 16 of the UFSAR which are consistent with those established in ITS 5.5.1 for the ODCM and is consistent with the CTS.

16 This change adds ITS 5.5.2 which reflects the NRC model for implementation of Option B to 10 CFR, Appendix J, enclosed in a letter from C. I. Grimes, NRC, to D. J. Modeen, NEI, dated November 2, 1995.

17 CTS 6.1.1.6 requirements for the manpower necessary to achieve alternate shutdown are retained in ITS 5.2.2.h. These requirements were reviewed and approved by the NRC by issuance of amendment 195/195/192 on 5/11/92.

18 Not used.

19 Not used.

20 NUREG 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," is not adopted. ONS will maintain the RCS Pressure and Temperature Curves and Limits in the ITS and will not implement a PTLR at this time.

21 Approved TSTF-118 is not adopted for NUREG 5.5.9. TSTF-118 specifies that SR 3.0.2 is applicable to the Steam Generator Tube Surveillance Program testing frequencies. SR 3.0.2 permits a 25% interval extension for performing SRs. The Steam Generator Tube Surveillance Program 5.5.10.d specifies Inspection Intervals with a minimum and maximum interval (e.g., " . . . intervals of not less than 12 months nor more than 24 months"). The application of the provisions of SR 3.0.2 is not consistent with this manner of specifying testing frequencies. The

ITS Section 5.0

ID 88

Subject: Revised ITS 5.4.1.b to include reference to NUREG-0737...

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

2 b. The emergency operating procedures required to implement the
2 requirements of NUREG-0737 and NUREG-0737, Supplement 1, as
2 stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. Fire Protection Program Implementation; and

e. All programs specified in Specification 5.5.

Specification 5.0

A1 <except as marked>

6.4 STATION OPERATING PROCEDURES

Written procedures shall be established implemented & maintained covering the following activities
a. The applicable procedures recommended in RG 1.33, Rev 2, App. A, Feb. 1978

Specification

6.4.1 The station shall be operated and maintained in accordance with approved procedures. Written procedures with appropriate check-off lists and instructions shall be provided for the following conditions:

5.4.1
5.4.1a

a. Normal startup, operation, and shutdown of the complete facility and of all systems and components involving nuclear safety of the facility. — A9

b. Refueling operations.

required to implement the requirements of NUREG-0737 and NUREG-0737 Supplement 1, as stated in Generic Letter 82-33, — A42

Supp 2

c. Actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety and radiation levels, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

5.4.1b

d. Emergency operating procedures involving potential or actual release of radioactivity.

5.4.1.a

e. Preventive or corrective maintenance which could affect nuclear safety or radiation exposure to personnel. — A9

f. Station survey following an earthquake. — LA10

5.4.1.a

g. Personnel radiation protection procedures.

h. Operation of radioactive waste management systems. — A9

i. Control of Ph in recirculated coolant after loss-of-coolant accident. Procedure shall state that pH will be measured and the addition of appropriate caustic to coolant will commence within 30 minutes after switchover to recirculation mode of core cooling to adjust the pH to a range of 7.0 to 8.0 within 24 hours. — LA12

5.4.1.a

j. Nuclear safety-related periodic test procedures. — A9

A41 CTS 3.10.1 requires limiting the radioactive quantity in each waste gas holdup tank to ≤ 3.8E+05 Curies noble gas (considered as Xe-133). ITS 5.5.13.b requires the quantity in each gas storage tank to be less than the amount which would result in a whole body exposure of ≥ 0.5 rem to an individual at the nearest exclusion area boundary in the event of an uncontrolled release of the tank's contents. The CTS specified limit on tank contents represents a quantity which results on the ITS limit on whole body exposure. This is an administrative change and is consistent with the NUREG.

2 A42 CTS 6.4.1.d requires that written procedures with appropriate check-off
2 lists and instructions shall be provided for "emergency procedures
2 involving potential or actual release of radioactivity." ITS 5.4.1
2 provides comparable requirements with the exception of referring to
2 emergency operating procedures required to implement the requirements of
2 NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-
2 33. Since these requirements are equivalent, this is considered a
2 change in presentation only and is therefore administrative. The change
2 is consistent with the NUREG.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; 6.4.1a,b,c,e, g,h,i,j,k,l

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33]. (1) 6.4.1.d

Supp 2

c. Quality assurance for effluent and environmental monitoring; Doc M4

Supp. 1

d. Fire Protection Program implementation; and

e. All programs specified in Specification 5.5. Doc M11

11 NUREG 5.2.2.e is modified to retain the current licensing basis regarding controls to limit working hours. Additionally, the term "auxiliary operator" is replaced with "non-licensed operator." This change is editorial to reflect current plant terminology.

12 Plant specific terminology is utilized in ITS 5.2.2.g for clarity. The shift work manager provides the accident assessment function. This is consistent with the current licensing basis.

2 13 NUREG 5.3.1 is modified to retain exceptions to qualification
2 requirements for the Operations Superintendent and the Shift Operations
2 Manager.

2 14 Not used.

15 The Radiological Effluent Technical Specifications were previously relocated to Chapter 16 of the UFSAR, rather than the ODCM. Section 5.5.1.b of the NUREG which addresses these controls has been included in ITS 5.5.5. ITS 5.5.5 includes controls for chapter 16 of the UFSAR which are consistent with those established in ITS 5.5.1 for the ODCM and is consistent with the CTS.

16 This change adds ITS 5.5.2 which reflects the NRC model for implementation of Option B to 10 CFR, Appendix J, enclosed in a letter from C. I. Grimes, NRC, to D. J. Modeen, NEI, dated November 2, 1995.

17 CTS 6.1.1.6 requirements for the manpower necessary to achieve alternate shutdown are retained in ITS 5.2.2.h. These requirements were reviewed and approved by the NRC by issuance of amendment 195/195/192 on 5/11/92.

18 Not used.

19 Not used.

20 NUREG 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," is not adopted. ONS will maintain the RCS Pressure and Temperature Curves and Limits in the ITS and will not implement a PTLR at this time.

21 Approved TSTF-118 is not adopted for NUREG 5.5.9. TSTF-118 specifies that SR 3.0.2 is applicable to the Steam Generator Tube Surveillance Program testing frequencies. SR 3.0.2 permits a 25% interval extension for performing SRs. The Steam Generator Tube Surveillance Program 5.5.10.d specifies Inspection Intervals with a minimum and maximum interval (e.g., " . . . intervals of not less than 12 months nor more than 24 months"). The application of the provisions of SR 3.0.2 is not consistent with this manner of specifying testing frequencies. The

ITS Section 5.0

ID 89

Subject: Revised ITS 5.5.3 to include identification of systems that have portions of system outside containment that could include highly radioactive fluids.

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall
2 be 0.25% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests;

2 b. Leakage > 0.50 L_a shall be to the penetration room.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.3 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to
2 levels as low as practicable. These systems include High Pressure
2 Injection, Low Pressure Injection, Reactor Building Spray, Gaseous
2 Waste Disposal, Makeup and Purification, Chemical Addition and
2 Sampling, and Coolant Treatment. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

Unit 1

Specification 5.0

(A1) (except as marked)

Secondary Water Chemistry

5.5.11 G. The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for the parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions of off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

5.5.3 H. ~~Systems Integrity~~ Reactor Coolant Sources Outside Containment

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

Supp 2

INSERT CTS CONDITION H - M20

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

5.5.4 I. ~~Iodine Monitoring~~ Post Accident Sampling

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

Insert CTS Condition H

These systems include High Pressure Injection, Low Pressure Injection, Reactor Building Spray, Gaseous Waste Disposal, Makeup and Purification, Chemical Addition and Sampling, and Coolant Treatment.

Unit 2 (A1) <except as marked> Specification 5.0

Secondary Water Chemistry

G. The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include: 5.5.1.1

1. Identification of a sampling schedule for the critical parameters and control points for the parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions of off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

H. ~~Systems Integrity~~ Primary Coolant Sources Outside Containment 5.5.3

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following: Supp 2

INSERT LTS CONDITION H (M20)

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

I. ~~Iodine Monitoring~~ Post Accident Sampling 5.5.4

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

Insert CTS Condition H

These systems include High Pressure Injection, Low Pressure Injection, Reactor Building Spray, Gaseous Waste Disposal, Makeup and Purification, Chemical Addition and Sampling, and Coolant Treatment.

Specification 5.0

Unit 3

① (except as marked)

-9-

Secondary Water Chemistry

5.5.11

G. The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1. Identification of a sampling schedule for the critical parameters and control points for the parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedure for the recording and management of data;

5. Procedures defining corrective actions of off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action. ⑧

H. ~~Systems Integrity~~ Primary Coolant Sources Outside Containment

5.5.3

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

Supp 2

INSERT CTS CONDITION H — M20

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

I. Iodine Monitoring Post Accident Sampling

5.5.4

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

Insert CTS Condition H

These systems include High Pressure Injection, Low Pressure Injection, Reactor Building Spray, Gaseous Waste Disposal, Makeup and Purification, Chemical Addition and Sampling, and Coolant Treatment.

2 M18 CTS 6.1.1.4 states that minimum qualifications for station staff are encompassed by ANSI/ANS-3.1-1978 except the Radiation Protection Manager, the Operations Superintendent and the Shift Operations Manager. The second paragraph of CTS 6.1.1.4 specifies the minimum qualifications for the Radiation Protection Manager and the third paragraph allows the station manager to appoint someone of lesser qualifications provided it is based on recommendations from the Nuclear Technical Services Manager and approved by the Site Vice President. ITS 5.3.1 specifies that the Radiation Protection Manager meet the minimum qualifications of ANSI/ANS-3.1-1978. No exceptions are provided. The requirement is acceptable since it is reasonable and is not a significant burden on plant operations. The change is consistent with the NUREG.

2 M19 CTS Table 6.1-1, Additional Requirement 5 specifies that at least one person per shift shall have sufficient training to perform routine health physics requirements. ITS 5.2.2.d specifies that a Radiation Protection Technician shall be on site. Since the ITS only allows a Radiation Protection Technician to fulfill the previous requirement for the person on shift where before any individual with sufficient training to perform routine health physics requirements was adequate, the change is more restrictive. This more restrictive change is an acceptable restriction on plant operation and is consistent with the NUREG.

2 M20 License Condition H of Oconee Operating Licenses for Units 1, 2, and 3 requires the licensee to implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. ITS 5.5.3 provides comparable requirements and specifies the systems included in the program. Even though these systems are currently identified in the program that implements this requirement, specifying the systems is more restrictive since these systems were not previously identified by technical specifications as being required to be included. The change is consistent with the NUREG.

CTS

5.5 Programs and Manuals

DocM16

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

16 Insert 5.0-8A 4.4.1

5.5 (2)(3) Primary Coolant Sources Outside Containment

Lic. Cond 3.H

Gaseous Waste Disposal (1)

High Pressure Injection

Supp 2

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Low Pressure Injection, Reactor Building Spray, Makeup and Purification, ~~and Hydrogen Recombiner~~]. The program shall include the following:

(1)

Chemical Addition and Sampling, and Coolant Treatment

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

25

5.5 (3)(4) Post Accident Sampling

iodines

6.4.4.c
6.4.4.d

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive ~~gases~~, and particulates in plant gaseous effluents and containment atmosphere samples under accident conditions. The program shall include the following:

16

and airborne iodine concentrations in areas with

Lic. Cond 3.I

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

27

UFSAR Chapter 16

5.5 (4)(5) Radioactive Effluent Controls Program

6.4.4.e

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to

(continued)

inclusion of this change can create confusion since the change can be construed to permit exceeding the maximum specified interval. Accordingly, the CLB regarding Steam Generator Tube Surveillance testing intervals is retained without adopting TSTF-118.

22 Approved TSTF-118 is not adopted for NUREG 5.5.13. TSTF-118 specifies that SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies. SR 3.0.2 permits a 25% interval extension for performing SRs. SR 3.0.3 permits up to a 24 hour extension to declare the LCO not met if an SR is discovered to not have been performed within its specified Frequency. The testing Frequency for DG fuel oil is specified in ITS 3.10.1 Neither of these changes are appropriate since the CLB adopted for the Diesel Fuel Oil Testing Program does not specify a Frequency for fuel oil testing. Accordingly, the CLB regarding the Diesel Fuel Oil Testing Program is retained without adopting TSTF-118.

23 Bracketed reviewers Note is deleted.

2 24 Not used.

25 The CTS 6.4.4.d requirements specifying the capability to obtain and analyze radioactive iodine in plant gaseous effluent is retained in ITS 5.5.4. These requirements were reviewed and approved by the NRC by issuance of amendment 211/211/208 on 12/1/95.

26 License Condition 3.I specifying the requirement to obtain and analyze airborne iodine in vital areas, is incorporated into ITS 5.5.4. This requirement was reviewed and approved by the NRC by issuance of amendment 92/92/89 on 1/28/81.

27 The CTS 6.4.4.e requirement regarding the location of the program for control of radioactive effluents and maintaining doses the MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable is retained in ITS 5.5.5. This requirement was reviewed and approved by the NRC by issuance of amendment 211/211/208 on 12/1/95.

28 CTS 6.4.4.e.2 requirements regarding limitations on the concentrations of radioactive materials released in liquid effluents to unrestricted is retained in ITS 5.5.5.b. These requirements were reviewed and approved by the NRC by issuance of amendment 211/211/208 on 12/1/95.

29 CTS 6.4.4.e.7 requirements regarding limitations on the concentrations of radioactive materials released in gaseous effluents to areas beyond the site boundary is retained in ITS 5.5.5.g. These requirements were reviewed and approved by the NRC by issuance of amendment 211//208 on 12/1/95.

30 CTS 4.2.3 requirements are retained in lieu of the NUREG 5.5.7 requirements. ONS is not committed to Regulatory Guide 1.14 specified in the NUREG for Reactor Coolant Pump Flywheel Inspections.

31 The CTS 4.17 requirements regarding the Steam Generator Tube Surveillance Program are retained in ITS 5.5.10 in accordance with the NUREG Reviewer's Note. These requirement were reviewed and approved by the NRC by issuance of amendment 161/161/158 on 9/1/87.

32 The NUREG 5.5.10 requirements regarding monitoring of secondary water chemistry associated with inhibiting low pressure turbine disc stress corrosion cracking is not adopted in ITS since the CTS does not include comparable requirements. The License Condition 3.G requirements monitoring secondary water chemistry are retained. This requirement was reviewed and approved by the NRC by issuance of amendment 92/92/89 on 1/28/81.

33 The CTS 4.12 requirements for testing the filters of the Control Room Pressurization and Filtering System and the CTS 4.5.4 requirements for testing the filters of the Penetration Room Ventilation System are retained in ITS 5.5.12. The CTS requirements for testing the filters of the Control Room Pressurization and Filtering System were reviewed and approved by the NRC by issuance of amendment 174/174/171 on 6/6/89. The CTS requirements for testing the filters of the Penetration Room Ventilation System were reviewed and approved by the NRC by issuance of amendment 215/215/212 on 4/2/96.

34 The CTS 3.9.1.b requirements associated with Explosive Gas and Storage Tank Radioactivity Monitoring is retained in ITS 5.5.13. These requirements were reviewed and approved by the NRC by issuance of amendment 187/187/184 on 1/22/91.

35 The CTS 4.20.3.a.3 requirements for Diesel Fuel Oil Testing are retained in ITS 5.5.14. NRC approved TSTF-106 is not adopted. TSTF-106 clarifies requirements regarding testing requirements for new fuel following addition to the storage tanks. This change is not adopted since the CLB does not include testing the associated fuel oil properties. These requirements were reviewed and approved by the NRC by issuance of amendment 195/195/192 on 5/11/92.

36 License Condition 3.J specifying the backup capability for determining subcooling margin is retained in ITS 5.5.17. This requirement was reviewed and approved by the NRC by issuance of amendment 92/92/89 on 1/28/81.

37 ITS 5.6.2 requirements regarding the contents of the Annual Radiological Environmental Operating Report is not adopted since there is no

comparable CTS requirement. These requirements were reviewed and approved by the NRC by issuance of amendment 187/187/184 on 1/22/91.

38 The NUREG 5.6.4 requirement regarding inclusion of documentation for challenges to pressurizer power operated relief valves or safety valves in the Monthly Operating Report is not adopted since there is no comparable CTS requirement. By correspondence dated 7/16/84, Duke Power committed to report such failures as Licensee Event Reports.

39 The bracketed NUREG 5.6.7 requirements regarding the EDG Failure Report is not adopted since there is no comparable CTS requirements.

40 Not used.

41 The CTS 4.17 requirements regarding the Steam Generator Tube Inspection Report are retained in ITS 5.6.8 in accordance with the Reviewer's Note.

42 The bracketed NUREG 5.7 is not adopted since there is no comparable CTS requirement.

43 Bracketed NUREG 5.2.2.e is modified to retain current requirements regarding periodic review of individual overtime.

44 NUREG Specification 5.5.8, "Inservice Testing (IST) Program," is modified to state that the IST Program provides control for ASME Code Class 1, 2, and 3 "pumps and valves," in place of "components including applicable supports." 10 CFR 50.55a(f) provides the regulatory requirements for an IST Program, and specifies that ASME Code Class 1, 2, and 3 pumps and valves are the only components covered by an IST Program. 10 CFR 50.55a(g) provides regulatory requirements for an Inservice Inspection (ISI) Program, and specifies that ASME Code Class 1, 2, and 3 components (including supports) are covered by the ISI Program, and that pumps and valves are covered by the IST Program in 10 CFR 50.55a(f). The ISTS does not include ISI Program requirements, as these program requirements have been relocated to plant specific documents. Therefore, the "applicable support" requirements are deleted and the components the IST Program applies to (i.e., pumps and valves) are added for clarity. Additionally, the statement, "The Program shall include the following:" is deleted since not all the statements that follow are really part of the program requirements.

45 Nureg 5.5.12.b is modified to require gas storage tank quantities be less than an amount which would result in a whole body exposure of 0.5 rem to an individual at the nearest exclusion are boundary in lieu of an individual in an unrestricted area. The CLB as indicated in the Bases to CTS 3.10 limits tank contents based on potential whole body exposure to an individual at the nearest exclusion area boundary. This change is

necessary since the 10 CFR 20 definition for restricted area includes some areas within the ONS exclusion area. This provision was approved by amendments 187/187/184 issued on 1/22/91.

46 NUREG 5.5.10.c is modified to delete the provisions regarding monitoring the discharge of the condensate pumps. Specific sampling requirements are maintained under administrative control which provides greater flexibility in establishing monitoring requirements based on plant specific experience. The Secondary Water Chemistry Monitoring Program requirements were approved by the NRC by issuance of amendments 83/83/80 on 6/12/80.

ITS Section 5.0

ID 90

Subject: Revised ITS 5.5.8 to retain lead-in sentence: "This program shall provide for the inspection of each reactor coolant pump flywheel."

5.5 Programs and Manuals (continued)

5.5.6 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR, Section 5.2.1.4, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.7 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1990.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.8 Reactor Coolant Pump Flywheel Inspection Program

2 This program shall provide for inspection of each reactor coolant
2 pump flywheel. At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an inplace, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed if the interval measured from the previous such inspection is greater than 6 2/3 years. The interval may be extended up to one year to permit inspections to coincide with a planned outage.

5.5 Programs and Manuals

5.5(4) Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

6.4.4.e.9

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

6.4.4.e.10

(15) Insert 5.0-10A 1.8.2

5.5(5) Component Cyclic or Transient Limit

(4) (4)

This program provides controls to track the FSAR, Section * *, cyclic and transient occurrences to ensure that components are maintained within the design limits.

Doc M3

5.2.1.4 (1)

5.5(6) Pre-Stressed Concrete Containment Tendon Surveillance Program

4.4.2.1

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~[~~Regulatory Guide 1.35, Revision 3, 1989~~]~~

(1)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

Docs Doc M5

5.5(X) Reactor Coolant Pump Flywheel Inspection Program

4.2.3

Supp 2

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

(30) Insert 5.0-10B

(continued)

ITS Section 5.0

ID 91

Subject: Revised ITS 5.5.10 to include the following lead-in : "This program provides the controls for SG tube surveillance. The program shall include the following:"

5.5 Programs and Manuals (continued)

5.5.10 Steam Generator (SG) Tube Surveillance Program

2 2 This program provides the controls for SG tube surveillance. The program shall include the following:

a. Examination Methods

Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

b. Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

2 2 2 2 2 2 2 2 2 2 2 For Units 1 and 3, there are a number of steam generator tubes which exceed the tube repair limit as a result of tube end anomalies. These tubes are temporarily exempted from the requirements for sleeving, rerolling or removal from service, until repaired during or before the next Unit 1 and Unit 3 refueling outages (Unit 1 EOC 18, Unit 3 EOC 17 refueling outages, respectively). An analysis has been performed which confirms the operability of Units 1 and 3 will not be impacted with these tubes in service until the next refueling outage on each of these units.

c. Selection and Testing

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.10-1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.10.d and the inspected tubes shall be verified acceptable per 5.5.10.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

(continued)

INSERT 50-11A

5.5.10 Steam Generator (SG) Tube Surveillance Program

2 This program provides the controls for SG tube surveillance. The
2 program shall include the following:

a. Examination Methods

Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

b. Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

2 For Units 1 and 3, there are a number of steam generator
2 tubes which exceed the tube repair limit as a result of tube
2 end anomalies. These tubes are temporarily exempted from
2 the requirements for sleeving, rerolling or removal from
2 service, until repaired during or before the next Unit 1 and
2 Unit 3 refueling outages (Unit 1 EOC 18, Unit 3 EOC 17
2 refueling outages, respectively). An analysis has been
2 performed which confirms the operability of Units 1 and 3
2 will not be impacted with these tubes in service until the
2 next refueling outage on each of these units.

c. Selection and Testing

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.10-1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.10.d and the inspected tubes shall be verified acceptable per 5.5.10.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

1. The first sample inspection during each inservice inspection of each steam generator shall include:

ITS Section 5.0

ID 92

Subject: Revised ITS 5.5.10 to indent appropriately.

5.5 Programs and Manuals (continued)

5.5.10 Steam Generator (SG) Tube Surveillance Program

2 2 This program provides the controls for SG tube surveillance. The program shall include the following:

a. Examination Methods

Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

b. Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

2 2 2 2 2 2 2 2 2 2 For Units 1 and 3, there are a number of steam generator tubes which exceed the tube repair limit as a result of tube end anomalies. These tubes are temporarily exempted from the requirements for sleeving, rerolling or removal from service, until repaired during or before the next Unit 1 and Unit 3 refueling outages (Unit 1 EOC 18, Unit 3 EOC 17 refueling outages, respectively). An analysis has been performed which confirms the operability of Units 1 and 3 will not be impacted with these tubes in service until the next refueling outage on each of these units.

c. Selection and Testing

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.10-1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.10.d and the inspected tubes shall be verified acceptable per 5.5.10.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

1. The first sample inspection during each inservice inspection of each steam generator shall include:

   a. All tubes that previously had detectable wall penetrations (>20%) and have not been plugged or sleeve repaired in the affected area.

   b. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

   c. A tube adjacent to any selected tube which does not permit passage of the eddy-current probe for tube inspection.

2. Tubes in the following Group(s) may be excluded from the first sample if all tubes in a Group in both OTSG are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

   Group A-1: Tubes within one, two, or three rows of the open inspection lane.

2 3. All tubes which have been repaired using the reroll process
2 will have the new roll area inspected during the inservice
2 inspection.
2
2 4. The tubes selected as the second and third samples (if required by Table 5.5.10-1) during each inservice inspection may be subjected to less than a full tube inspection provided:

   a. The tubes selected for these samples include the tubes from those areas of the tubesheet array where tubes with imperfections were previously found.

   b. The inspections include those portions of the tubes where imperfections were previously found.

   The results of each sample inspection shall be classified into one of the following three categories:

(continued)

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but no more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 5.5.10.c.2, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection, unless the mechanism of degradation is random in nature.

2 (3) Where special inspections are performed pursuant
2 to 5.5.10.c.2, defective or degraded tube
2 indications found in the new roll area as a
2 result of the inspection and any indications
2 found in the originally rolled region of the
2 rerolled tube, need not be included in
2 determining the Inspection Results Category for
2 the general steam generator inspection.

(continued)

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

d. Inspection Intervals

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies.

1. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of 40 months.

2. If the results of the inservice inspection of a steam generator performed in accordance with Table 5.5.10-1 at 40 month intervals fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 months nor more than one fuel cycle after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 5.5.10.d.1 and the interval can be extended to a maximum of 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.10-1 during the shutdown subsequent to any of the following conditions:

   a. A seismic occurrence greater than the Operating Basis Earthquake,

   b. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   c. A main steam line or feedwater line break.

(continued)

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

4. After primary to secondary leakage in excess of the limits of Specification 3.4.13, an inspection of the affected steam generator will be performed in accordance with the following criteria:

   a. If the leaking tube is in a Group as defined in Section 5.5.10.c.2, all of the tubes in this Group in this steam generator will be inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

2 b. If the leaking tube has been repaired by the reroll
2 process and is leaking in the new roll area, all tubes in
2 the steam generator that have been repaired by the reroll
2 process will have the new roll area inspected. If he
2 results of this inspection fall into the C-3 category,
2 additional inspections will be performed in the new roll
2 area in the other steam generator.
2
2 c. If the leaking tube is not in a Group as defined in 5.5.10.d.4.a, then an inspection will be performed on the affected steam generator in accordance with Table 5.5.10-1 with an initial inspection sample size of 6% of the tubes in the affected steam generator.

e. Definitions

As used in this specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube or a sleeve.

(continued)

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

3. Degraded Tube means a tube or a sleeve containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube or sleeve containing a defect is defective.

6. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

   The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

2 The rerolling repair process will only be used to repair
2 tubes with defects in the upper tubesheet area. The
2 rerolling repair process will be performed only once per
2 steam generator tube using a 1 inch reroll length. The new
2 roll area must be free of degradation in order for the
2 repair to be considered acceptable. The rerolling process
2 used by Oconee is described in the Topical Report, BAW-2303,
2 Revision 3.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.10.d.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

(continued)

TABLE 5.5.10-1 (Page 1 of 2)
STEAM GENERATOR TUBE INSPECTION

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this SG.	C-1	None
					C-2	Plug or repair defective tubes.
					C-3	Plug or repair defective tubes and perform action for C-3 result of 1st Sample.
			C-3	Plug or repair defective tubes and perform actions for C-3 results on 1st Sample.	N/A	N/A

(continued)

TABLE 5.5.10-1 (Page 2 of 2)
STEAM GENERATOR TUBE INSPECTION

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
(continued)	C-3	Inspect 6S tubes in the S.G, plug or repair defective tubes and inspect 2S tubes in the other S.G. Perform follow-on inspections in the other S.G. in accordance with results of the above inspection as applied to Table 5.5.10-1 Prompt Notification to NRC pursuant to 10 CFR 50.72	C-1	N/A	N/A	N/A
			C-2	N/A	N/A	N/A
			C-3 (2)	(a) If defects can be localized to an affected area, inspect all tubes in affected area and plug or repair defective tubes. (b) If defects cannot be localized to an affected area, inspect all tubes in this S.G. and plug or repair defective tubes.	C-1 C-2 C-3	N/A N/A N/A

Notes: (1) S=3(N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

(2) Following an 18% random inspection (C-3 category inspection) an unaffected area is identified. The unaffected area will be logically and consistently defined based on generator design, defect location and characteristics. The criteria for accepting an area as unaffected depends on the number of defects found in the sample inspected in that area and are established such that there is a 0.05 or smaller probability of accepting the area as unaffected if it contains 30 or more defective tubes.

ITS Section 5.0

ID 93

Subject: Revised ITS 5.5.13 to be more explicit - add: "...explosive gas mixtures and the quantity of radioactivity in gas storage tanks, and the quantity of radioactivity in unprotected..."

5.5 Programs and Manuals

5.5.12 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate, for the PRVS, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

d. Demonstrate, for the CRVS Booster Fan Trains, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

e. Demonstrate, for the CRVS Booster Fan Trains and PRVS, that a laboratory test of a sample of the carbon adsorber shows ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).

f. Demonstrate, for the PRVS, that the pressure drop across the combined HEPA filters and carbon adsorber banks is less than 6 in. of water at the system design flow rate ± 10%.

g. Demonstrate, for the CRVS Booster Fan Trains, that the pressure drop across the pre-filter is ≤ 1 in. of water and the pressure drop across the HEPA filters is ≤ 2 in. of water at the system design flow rate ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas
2 mixtures contained in the waste gas holdup tanks and the quantity
2 of radioactivity contained in waste gas holdup tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined. The liquid radwaste quantities shall be determined by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

(continued)

5.5 Programs and Manuals

5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limit for concentration of hydrogen in the waste gas holdup tanks and a surveillance program to ensure the limit is maintained. The limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); 2

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas holdup tank is less than the amount that would result in a whole body exposure of $\geq$ 0.5 rem to any individual at the nearest exclusion area boundary, in the event of an uncontrolled release of the tank's contents. 2

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than 10 curies excluding tritium and dissolved or entrained gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.14 Standby Shutdown Facility (SSF) Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of SSF fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the acceptability of Day Tank and Underground Storage Tank fuel oil for use by determining that the fuel oil viscosity, water and sediment are within limits.

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

ESF Ventilation System	Delta P	Flowrate
[ ]	[ ]	[ ]

e. [Demonstrate that the heaters for each of the ESF systems dissipate the value specified below [± 10%] when tested in accordance with [ASME N510-1989].]

ESF Ventilation System	Wattage
[ ]	[ ]

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

[5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the [waste gas holdup system], [the quantity of radioactivity contained in gas storage tanks ~~or fed into the offgas treatment system~~], and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in [Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"]. The liquid radwaste quantities shall be determined in accordance with [Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"].

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the [waste gas holdup system] and a surveillance program to ensure the limits are maintained. Such limits shall be]

(continued)

5.5 Programs and Manuals

5.5.18 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank ~~and fed into the offgas treatment system~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual ~~in an unrestricted area~~, in the event of an uncontrolled release of the ~~tanks~~ contents;

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not ~~have tank overflows and surrounding area drains connected to the~~ Liquid Radwaste Treatment System, is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.19 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of ~~both new fuel oil and stored~~ fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish ~~the following:~~

~~a.~~ ~~Acceptability of new~~ fuel oil for use ~~prior to addition to storage tanks~~ by determining that the fuel oil ~~has~~

~~b.~~ an API gravity or an absolute specific gravity within limits,

(continued)

ITS Section 5.0

ID 94

Subject: Revised ITS 5.6.5 and 5.6.8 to include lead-in sentences.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6.3 Radioactive Effluent Release Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

2 Core operating limits shall be established, determined and issued
2 in accordance with the following:

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload

(continued)

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

2 5.6.8 Steam Generator Tube Inspection Report

2 The steam generator tube inspection report shall comply with the
2 following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes plugged or repaired.

2 4. Number of tubes repaired by rerolling and number of
2 indications detected in the new roll area of the
2 repaired tubes

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

6.6.1.2

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. (38)

Supp 2

Core operating limits shall be established, determined and issued in accordance with the following: (2) (1)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following: 6.9.1

Insert 5.0-20A

The individual specifications that address core operating limits must be referenced here.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: 6.9.2

Insert 5.0-20B

Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date. (1)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met. 6.9.3

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC. 6.9.4

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (20)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic

(continued)

5.6 Reporting Requirements (continued)

5.6 (9) (7)

Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

3.6.7.1.c
3.6.7.2.c
6.6.3.F

Supp 2

5.6 (10) (8)

Steam Generator Tube Inspection Report (10a) (2)

4.17.6

Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.

Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

INSERT 5.0-23A

41

INSERT 5.0-23A

2 The steam generator tube inspection report shall comply with the following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 and require notification to the NRC pursuant to 10 CFR 50.72 shall be reported prior to resumption of plant operation. The written followup of this report shall provide the results of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The designation of affected and unaffected areas will be reported to the NRC when they are determined.

ITS Section 5.0

ID 114

Subject: Fixed indentation problem on page 5.0-13 of ITS.

5.5 Programs and Manuals (continued)

5.5.9 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

2 b. The provisions of SR 3.0.2 are applicable to the above
2 required Frequencies for performing inservice testing
2 activities;
2
2 c. The provisions of SR 3.0.3 are applicable to inservice
2 testing activities; and
2
2 d. Nothing in the ASME Boiler and Pressure Vessel Code shall be
2 construed to supersede the requirements of any TS.

2
2
2
2
2
2

ITS Section 5.0

ID 115

Subject: Section 5.0 DOC M9 revised to refer to ITS 5.5.12.e instead of 5.5.12.c.1.

that ITS SR 3.0.3 is applicable to the Explosive gas and Radioactivity Monitoring Program does not exist.

A CTS requirement comparable to ITS SR 3.0.3 does not exist (refer to DOC M2 for ITS Section 3.0). ITS SR 3.0.3 requires a missed surveillance be performed within 24 hours or the LCO declared not met. CTS 4.0.1 does not explicitly require satisfactory completion of a surveillance within its specified frequency. Since an explicit connection between the performance of a surveillance within the specified frequency and compliance with the LCO does not exist, the CTS does not impose an explicit time limit to complete the missed surveillance. The addition of this explicit administrative requirement is reasonable and is not a significant impact upon unit operations. The explicit time limit to complete a missed SR is a more restrictive requirement upon plant operation and is consistent with the NUREG.

M6 CTS limits upon RCS Pressure, Temperature, and Flow to prevent departure from nucleate boiling (DNB) limits do not exist. These limits are added to the COLR. Limits on RCS pressure, temperature, and flow rate are provided "to ensure that the core operates within the limits assumed for the plant safety analyses." Operating within these limits will result in meeting departure from nucleate boiling ratio (DNBR) criteria in the event of a DNB limited transient. The addition of these limits to the COLR is a more restrictive requirement upon unit operation.

M7 Not used.

M8 Upon discovery of abnormal degradation of the containment structural integrity, CTS 3.6.7.1.c and CTS 3.6.7.2.c require submittal of a special report to the NRC. The scope and content of the special report is not specified. In this condition, ITS 5.6.7 requires submittal of a Tendon Surveillance Report. ITS requires the Tendon Surveillance report to include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken. The more prescriptive requirements in ITS are more restrictive requirements upon unit operation and are consistent with the NUREG. The administrative requirements regarding the scope and content for the Tendon Surveillance Report are reasonable and do not represent a significant impact upon unit operations.

2 M9 A CTS requirement comparable to ITS 5.5.12.e does not exist. ITS
2 5.5.12.e requires that a laboratory sample of the charcoal adsorber shows the radioactive methyl iodide removal capability for each Control Room Ventilation System Booster Fan Trains is within limit. This is a more restrictive requirement upon unit operations. Periodic

verification of radioactive iodine removal capability provides reasonable assurance that the removal capability is consistent with the assumptions of the design analysis for the Control Room Ventilation System.

M10 CTS 6.1.1.9 states the shift manager is an experienced SRO, who has been instructed in additional academic subjects, and will be assigned on-shift to provide the accident assessment capability. ITS 5.2.2.g states the Shift Work Manager, whose functions include those of a Shift Technical Advisor (STA), shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. In addition, the Shift Work Manager shall meet the qualifications for STA specified by the Commission Policy Statement on Engineering Expertise on Shift. The ITS requirements are more prescriptive and therefore a more restrictive requirement upon unit operations. The additional requirements are acceptable since they are reasonable and are consistent with existing practice.

M11 CTS Specification 6.4.1 requires written procedures for specified activities and conditions. ITS 5.4.1.e requires that each program specified in ITS Section 5.5 have written procedures established, implemented and maintained. Specifically, those programs for which this is a new requirement are 1) Component Cyclic or Transient Limits, 2) Technical Specification Bases Control Program, and 3) Safety Function Determination Program. This change imposes new requirements which are more restrictive, and has no adverse impact on safety. The requirement to establish, implement and maintain procedures for these new programs is reasonable and is not a significant impact upon unit operation.

M12 Explicit CTS requirements regarding licensee initiated changes to the ODCM do not exist. ITS 5.5.1 establishes explicit requirements for such changes. Therefore, this change is a more restrictive requirement upon unit operations. The addition of these requirements is acceptable since it is reasonable, generally consistent with current practice and not a significant impact upon unit operations.

M13 Explicit CTS requirements regarding licensee initiated changes to the Radiological Effluent Controls of the UFSAR do not exist. ITS 5.5.5 establishes explicit requirements for such changes. Therefore, this change is a more restrictive requirement upon unit operations. The addition of these requirements is acceptable since it is reasonable, generally consistent with current practice and not a significant impact upon unit operations.

ITS Section 5.0

ID 116

Subject: Revised Section 5.0 DOC LA15 to say that changes to QA Topical are controlled by...

LA14 CTS 6.4.4.f, Radiological Environmental Monitoring Program, requirements are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA15 CTS 6.5 requirements for records retention are relocated to the Quality Assurance Topical Report. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the QA Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA16 CTS 6.6.1.1 contains requirements for submitting a report following receipt of an operating license; installation of fuel that has a different design or has been manufactured by a different fuel supplier; modifications that may have altered the nuclear, thermal, or hydraulic performance of the unit; and amendments to the license involving planned increase in power operation. The change relocates these requirements to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA17 CTS 6.1.1.9 requires the Shift Manager, who provides the on-shift accident assessment capability, to be an experienced SRO. The requirement for this individual to be an experienced SRO is relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since

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Subject: Revised Section 5.0 DOC LA20 to indicate what controls changes for relocated info.

relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA18 CTS 4.5.5 requires LPI System leakage rate testing and provides an associated maximum allowable leakage limit. Its purpose is to maintain a preventive leakage rate for the LPI System which will prevent significant offsite exposures following an accident. This is one of the programs required by CTS License Condition H, which is addressed by ITS 5.5.3, "Primary Coolant Sources Outside Containment." The specific requirements of the LPI System leakage rate testing program are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA19 Not used.

LA20 CTS 4.4.2 provides details regarding testing associated with containment structural integrity. These details are relocated to the Containment Tendon Testing Program. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to the Containment Tendon Testing Program provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory
2 control and provides for a more appropriate change control process. The
2 level of safety of facility operation is unaffected by the change
2 because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore,
2 relocation of this detail is acceptable. This Program is subject to the
2 controls described in ITS Chapter 5 "Administrative Controls." This change is consistent with the NUREG.

LA21 CTS 4.5.4.1.c and 4.5.4.1.e provide details of the method of ventilation system filter testing which are not directly pertinent to the actual requirement. These details are relocated to UFSAR Chapter 16. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the details to UFSAR Chapter 16 provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA22 CTS 6.9.1.4 requires establishments of values and documentation in the COLR for limits associated with volume and boron concentration in the Concentrated Boric Acid Tank. These requirements are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

ITS Section 5.0

ID 129

Subject: Revised Section 5.0 DOC LA3 to refer only to CTS 4.12.2 (i.e., delete reference to CTS 4.12.1.a).

TECHNICAL CHANGES - REMOVAL OF DETAILS

LA1 CTS 3.9.1, 3.10.1 (and associated surveillance in CTS Table 4.1-3, Item 13) and 3.10.2 specify requirements for Liquid Holdup Tanks, Waste Gas Holdup Tanks, and Explosive Gas Mixtures, respectively. The details of the methods for implementing these requirements are relocated to UFSAR Chapter 16. A description of the Explosive Gas and Storage Tank Radioactivity Monitoring Program is incorporated into the Administrative Controls section of ITS which includes appropriate limits, actions and surveillance requirements. The details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the details to UFSAR Chapter 16 provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA2 TS 4.2.2 contains requirements associated with inspecting and maintaining the structural integrity of the reactor internals. These requirements are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

2 LA3 CTS 4.12.2 provide details of the method of ventilation system filter testing which are not directly pertinent to the actual requirement. These details are relocated to UFSAR Chapter 16. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the details to UFSAR Chapter 16 provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA4 CTS Table 6.1-1 specifies requirements for licensed and non-licensed personnel. This table specifies additional personnel than those required by 10 CFR 50.54(m) or ITS 5.2.2. The requirements for these additional personnel is relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA5 CTS 6.1.1.7 and 6.1.1.8 contain requirements regarding training for station personnel and the fire brigade. These training requirements are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA6 Not used.

LA7 CTS 6.1.2.1, 6.4.1.o and 6.4.1.p, which delineate the PORC and Technical Review and Control requirements; CTS 6.4.2 which requires each procedure required by 6.4.1 to be reviewed by an appropriate manager or designee; and CTS 6.4.3 which addresses temporary changes to procedures; are relocated to the Quality Assurance Topical Report. CTS 6.1.2.1 describes review and control requirements for procedures, modifications, tests, experiments, reportable events, special reviews and investigations, and unplanned onsite releases. CTS 6.1.2.1 also requires records of the above activities be provided as necessary for required reviews. These requirements are also located in ANSI N18.7-1976. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to the Quality Assurance Topical Report provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore,

NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the Quality Assurance Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA8 CTS 6.1.3 delineates the requirements of the offsite review committee (Nuclear Safety Review Board (NSRB)). The CTS requirements specify the Function, Organization, Review, Audits, and Responsibilities and Authorities of the NSRB. The proposed change moves these requirements to the Quality Assurance Topical Report. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to the Quality Assurance Topical Report provides reasonable assurance that these requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the QA Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA9 CTS 6.2.1 and 6.2.2 details associated with certain licensee internal actions to be taken for reportable events is moved to the Quality Assurance Topical Report. The regulatory requirements for submittal of the reports are contained in 10 CFR 50.73. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these details to the Quality Assurance Topical Report provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. Changes to the QA Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA10 CTS 6.4.1.f requires procedures for a station survey following an earthquake. This requirement is relocated to UFSAR Chapter 16. This requirement is not required to be in the ITS to provide adequate protection of the public health and safety since relocation of the requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced.

Therefore, relocation of this detail is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA11 CTS 6.4.1.n requires providing procedures for implementing the Process Control Program (PCP). This change relocates the requirements of CTS 6.4.1.n to the Quality Assurance Topical Report. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of the requirements to the Quality Assurance Topical Report provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the QA Topical Report are controlled by the provisions of 10 CFR 50.54. This change is consistent with the NUREG.

LA12 CTS 6.4.1.i includes details regarding the content of procedures for controlling ph in recirculated coolant after a LOCA. CTS 6.4.1.k includes details regarding the content of procedures for remote or local operation of components necessary to establish high and low pressure injection within 15 minutes of a line break. These details are relocated to UFSAR Chapter 16. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these details to UFSAR Chapter 16 provides reasonable assurance that the details are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these details are reduced. Therefore, relocation of these details is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

LA13 CTS 6.4.4.a requirements for a respiratory protective program are relocated to UFSAR Chapter 16. These requirements are not required to be in the ITS to provide adequate protection of the public health and safety, since relocation of these requirements to UFSAR Chapter 16 provides reasonable assurance that the requirements are implemented. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Furthermore, NRC and utility resources associated with processing license amendments to these requirements are reduced. Therefore, relocation of these requirements is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59. This change is consistent with the NUREG.

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ID 130

Subject: Revised Insert 5.0-20A to crossreference CTS 6.9.1.1 rather than CTS 2.1.

INSERT 5.0-20A

CTS

1. Shutdown Margin limit for Specification 3.1.1; DOC A35

2. Moderator Temperature Coefficient limit for Specification 3.1.3; DOC A35

3. Physical Position, Sequence and Overlap limits for Specification 3.2.1 Rod Insertion Limits, 6.9.1.3

4. AXIAL POWER IMBALANCE operating limits for Specification 3.2.2; 6.9.1.9

5. QUADRANT POWER TILT (QPT) limits for Specification 3.2.3; 6.9.1.8

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1; 6.9.1.2

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1; DOC M6

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1; 6.9.1.5

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4; 6.9.1.6

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12; 6.9.1.7

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and DOC A35

Supp. 2 12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1. 6.9.1.1

ITS Section 3.6 3.7 4.0 5.0

ID 139

Subject: Revised ITS to remove Penetration Room Ventilation System Tech Spec change 95-04

5.5 Programs and Manuals

5.5.2 Containment Leakage Rate Testing Program (continued)

This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall
2 be 0.25% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests;

2 b. Leakage > 0.50 L_a shall be to the penetration room.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.3 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to
2 levels as low as practicable. These systems include High Pressure
2 Injection, Low Pressure Injection, Reactor Building Spray, Gaseous
2 Waste Disposal, Makeup and Purification, Chemical Addition and
2 Sampling, and Coolant Treatment. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

Specification 5.0

(A1) (Except as Marked)

4.4 REACTOR BUILDING

4.4.1 Containment Leakage Tests

Applicability

Applies to Containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

ITS

Specification

This program provides controls for implementation of the

5.5.2

4.4.1.1 Integrated Leak Rate Tests

Testing of the containment

The containment leakage rate shall be determined, as required by 10CFR50.54 (o) and 10CFR50, Appendix J, Option B, including any approved exemptions, using the guidelines of Regulatory Guide 1.163, dated September, 1995.

as modified by

for Type A testing

4.4.1.1.1 Acceptance Criteria

maximum allowable

at Pa shall be ≤ 1.0

The program shall be in accordance with

0.50

5.5.2

Suppl. 2

The overall acceptance containment leakage rate is determined by the preoperational leakage rate test and shall not exceed L_a, 0.25 weight percent of containment air per 24 hours at P_a, 59 psig. Any leakage in excess of 50% of the total allowed containment leakage shall be demonstrated to be to the penetration room. Containment leakage prior to startup following a Type A test shall not exceed .75 L_a.

day

L_a

≤

following

15

Testing in accordance with this program

5.5.2

4.4.1.2 Local Leak Rate Testing

The peak calculated containment internal pressure for the Design Basis LOCA

4.4.1.2.1 Scope of Testing

Implemented in the program

Type B+C testing

The local leak rate shall be measured for the containment penetrations in accordance with the criteria specified in Appendix J of 10CFR50, Option A.

5.5.2

4.4.1.2.2 Frequency of Test

Local leak rate tests shall be conducted with gas at a pressure of not less than 59 psig during each reactor shutdown for refueling or other convenient interval but in no case at intervals greater than 24 months.

(A22)

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program

(M5)

Amendment No. 218 (Unit 1)
Amendment No. 218 (Unit 2)
Amendment No. 215 (Unit 3)

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Specification 5.0

(A1) (except as Marked)

During the first unit startup following testing in accordance with this program, the Leakage rate acceptance criterion (LA)

4.4.1.2.3 Acceptance Criteria for Type B & C tests

~~The combined leakage rate from all penetrations and isolation valves shall not exceed~~ ≤ 0.60 L_a ~~0.125 weight percent of the postulated post-accident containment air mass per 24 hours at 59 psig~~

Sulfur 2

4.4.1.3 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a local leak rate test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.1 and 4.4.1.2.3, respectively. (A2)

4.4.1.4 Isolation Valve Functional Tests

Inservice testing of ASME Code Class 1, 2, and 3 valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

(See 3.6)

Specification 5.0

CA21

Supp 2

c. Every 18 months, or following 720 hours of system operation, or after painting, fire, or chemical release in any ventilation zone communicating with the system, a carbon sample shall be removed from the Penetration Room Ventilation system filters for laboratory analysis. Within 31 days of removal this sample shall be verified to show ≥90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable. (See 3.7)

5.5.12.c

Supp 2

Bases

Pressure drop across the combined high efficiency particulate air (HEPA) filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per year operating cycle establishes performance capability.

(HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system every month will demonstrate operability of the filters and adsorber system. Operation for 15 minutes demonstrates operability and minimizes the moisture build up during testing.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

A2

30 of 68

INSERT 5.0-8A

5.5.2 Containment Leakage Rate Testing Program

This program provides controls for implementation of the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions for Type A testing. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A. 4.4.1.1 4.4.1.2.1

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 59 psig. 4.4.1.2.1

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% of the containment air weight per day. 4.4.1.1.1

2

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criteria is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests.

2 b. Leakage > 0.50 L_a shall be to the penetration room.

Doc MS

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

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ID 143

Subject: Incorporate TSC 98-07 (tube end anomalies) and Amendment 227/227/224 (SG reroll change)

5.5 Programs and Manuals (continued)

5.5.10 Steam Generator (SG) Tube Surveillance Program

2 This program provides the controls for SG tube surveillance. The
2 program shall include the following:

a. Examination Methods

Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

b. Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

2 For Units 1 and 3, there are a number of steam generator tubes
2 which exceed the tube repair limit as a result of tube end
2 anomalies. These tubes are temporarily exempted from the
2 requirements for sleeving, rerolling or removal from service,
2 until repaired during or before the next Unit 1 and Unit 3
2 refueling outages (Unit 1 EOC 18, Unit 3 EOC 17 refueling
2 outages, respectively). An analysis has been performed which
2 confirms the operability of Units 1 and 3 will not be impacted
2 with these tubes in service until the next refueling outage on
2 each of these units.

c. Selection and Testing

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.10-1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.10.d and the inspected tubes shall be verified acceptable per 5.5.10.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

1. The first sample inspection during each inservice inspection of each steam generator shall include:

   a. All tubes that previously had detectable wall penetrations (>20%) and have not been plugged or sleeve repaired in the affected area.

   b. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

   c. A tube adjacent to any selected tube which does not permit passage of the eddy-current probe for tube inspection.

2. Tubes in the following Group(s) may be excluded from the first sample if all tubes in a Group in both OTSG are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

   Group A-1: Tubes within one, two, or three rows of the open inspection lane.

2 3. All tubes which have been repaired using the reroll process
2 will have the new roll area inspected during the inservice
2 inspection.
2
2 4. The tubes selected as the second and third samples (if required by Table 5.5.10-1) during each inservice inspection may be subjected to less than a full tube inspection provided:

   a. The tubes selected for these samples include the tubes from those areas of the tubesheet array where tubes with imperfections were previously found.

   b. The inspections include those portions of the tubes where imperfections were previously found.

   The results of each sample inspection shall be classified into one of the following three categories:

(continued)

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but no more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 5.5.10.c.2, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection, unless the mechanism of degradation is random in nature.

2 (3) Where special inspections are performed pursuant
2 to 5.5.10.c.2, defective or degraded tube
2 indications found in the new roll area as a
2 result of the inspection and any indications
2 found in the originally rolled region of the
2 rerolled tube, need not be included in
2 determining the Inspection Results Category for
2 the general steam generator inspection.

(continued)

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

d. Inspection Intervals

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies.

1. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of 40 months.

2. If the results of the inservice inspection of a steam generator performed in accordance with Table 5.5.10-1 at 40 month intervals fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 months nor more than one fuel cycle after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 5.5.10.d.1 and the interval can be extended to a maximum of 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.10-1 during the shutdown subsequent to any of the following conditions:

   a. A seismic occurrence greater than the Operating Basis Earthquake,

   b. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   c. A main steam line or feedwater line break.

(continued)

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

4. After primary to secondary leakage in excess of the limits of Specification 3.4.13, an inspection of the affected steam generator will be performed in accordance with the following criteria:

   a. If the leaking tube is in a Group as defined in Section 5.5.10.c.2, all of the tubes in this Group in this steam generator will be inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

2 b. If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the new roll area in the other steam generator.

2 c. If the leaking tube is not in a Group as defined in 5.5.10.d.4.a, then an inspection will be performed on the affected steam generator in accordance with Table 5.5.10-1 with an initial inspection sample size of 6% of the tubes in the affected steam generator.

e. Definitions

As used in this specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube or a sleeve.

(continued)

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

3. Degraded Tube means a tube or a sleeve containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube or sleeve containing a defect is defective.

6. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

   The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

2 The rerolling repair process will only be used to repair
2 tubes with defects in the upper tubesheet area. The
2 rerolling repair process will be performed only once per
2 steam generator tube using a 1 inch reroll length. The new
2 roll area must be free of degradation in order for the
2 repair to be considered acceptable. The rerolling process
2 used by Oconee is described in the Topical Report, BAW-2303,
2 Revision 3.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.10.d.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

(continued)

TABLE 5.5.10-1 (Page 1 of 2)
STEAM GENERATOR TUBE INSPECTION

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this SG.	C-1	None
					C-2	Plug or repair defective tubes.
					C-3	Plug or repair defective tubes and perform action for C-3 result of 1st Sample.
			C-3	Plug or repair defective tubes and perform actions for C-3 results on 1st Sample.	N/A	N/A

(continued)

TABLE 5.5.10-1 (Page 2 of 2)
STEAM GENERATOR TUBE INSPECTION

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
(continued)	C-3	Inspect 6S tubes in the S.G, plug or repair defective tubes and inspect 2S tubes in the other S.G. Perform follow-on inspections in the other S.G. in accordance with results of the above inspection as applied to Table 5.5.10-1 Prompt Notification to NRC pursuant to 10 CFR 50.72	C-1	N/A	N/A	N/A
			C-2	N/A	N/A	N/A
			C-3 (2)	(a) If defects can be localized to an affected area, inspect all tubes in affected area and plug or repair defective tubes. (b) If defects cannot be localized to an affected area, inspect all tubes in this S.G. and plug or repair defective tubes.	C-1 C-2 C-3	N/A N/A N/A

Notes: (1) S=3(N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

(2) Following an 18% random inspection (C-3 category inspection) an unaffected area is identified. The unaffected area will be logically and consistently defined based on generator design, defect location and characteristics. The criteria for accepting an area as unaffected depends on the number of defects found in the sample inspected in that area and are established such that there is a 0.05 or smaller probability of accepting the area as unaffected if it contains 30 or more defective tubes.

5.5 Programs and Manuals (continued)

5.5.11 Secondary Water Chemistry

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.12 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

The VFTP is applicable to the Penetration Room Ventilation System (PRVS) and the Control Room Ventilation System (CRVS) Booster Fan Trains.

a. Demonstrate, for the PRVS, that a dioctyl phthalate (DOP) test of the high efficiency particulate air (HEPA) filters shows $\geq$ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate $\pm$ 10%.

b. Demonstrate, for the CRVS Booster Fan Trains, that a DOP test of the HEPA filters shows $\geq$ 99.5% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate $\pm$ 10%.

(continued)

5.5 Programs and Manuals

5.5.12 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate, for the PRVS, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

d. Demonstrate, for the CRVS Booster Fan Trains, that a halogenated hydrocarbon test of the carbon adsorber shows ≥ 99% removal when tested at in accordance with ANSI N510-1975 at the system design flow rate ± 10%.

e. Demonstrate, for the CRVS Booster Fan Trains and PRVS, that a laboratory test of a sample of the carbon adsorber shows ≥ 90% radioactive methyl iodide removal when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).

f. Demonstrate, for the PRVS, that the pressure drop across the combined HEPA filters and carbon adsorber banks is less than 6 in. of water at the system design flow rate ± 10%.

g. Demonstrate, for the CRVS Booster Fan Trains, that the pressure drop across the pre-filter is ≤ 1 in. of water and the pressure drop across the HEPA filters is ≤ 2 in. of water at the system design flow rate ± 10%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program

2 This program provides controls for potentially explosive gas
2 mixtures contained in the waste gas holdup tanks and the quantity of radioactivity contained in waste gas holdup tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined. The liquid radwaste quantities shall be determined by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

(continued)

5.5 Programs and Manuals

5.5.13 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limit for concentration of hydrogen in the waste gas holdup tanks and a surveillance program to ensure the limit is maintained. The limit shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); 2

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas holdup tank is less than the amount that would result in a whole body exposure of $\geq$ 0.5 rem to any individual at the nearest exclusion area boundary, in the event of an uncontrolled release of the tank's contents. 2

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than 10 curies excluding tritium and dissolved or entrained gases.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.14 Standby Shutdown Facility (SSF) Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of SSF fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the acceptability of Day Tank and Underground Storage Tank fuel oil for use by determining that the fuel oil viscosity, water and sediment are within limits.

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the updated UFSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR .

d. Proposed changes that meet the criteria of 5.5.15.b.1 or 5.5.15.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.16 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Safety Function Determination Program (SFDP) (continued)

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of safety function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.17 Backup Method for Determining Subcooling Margin

This program ensures the capability to accurately monitor the Reactor Coolant System Subcooling Margin. The program shall include the following:

a. Training of personnel, and

a. Procedures for monitoring.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescent dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Radiological Environmental Operating Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6.3 Radioactive Effluent Release Report

----------NOTE----------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

2 Core operating limits shall be established, determined and issued
2 in accordance with the following:

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin limit for Specification 3.1.1;

2. Moderator Temperature Coefficient limit for Specification 3.1.3;

3. Physical Position, Sequence and Overlap limits for Specification 3.2.1 Rod Insertion Limits;

4. AXIAL POWER IMBALANCE operating limits for Specification 3.2.2;

5. QUADRANT POWER TILT (QPT) limits for Specification 3.2.3;

6. Nuclear Overpower Flux/Flow/Imbalance and RCS Variable Low Pressure allowable value limits for Specification 3.3.1;

7. RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits for Specification 3.4.1

8. Core Flood Tanks Boron concentration limits for Specification 3.5.1;

9. Borated Water Storage Tank Boron concentration limits for Specification 3.5.4;

10. Spent Fuel Pool Boron concentration limits for Specification 3.7.12;

11. RCS and Transfer Canal boron concentration limits for Specification 3.9.1; and

12. AXIAL POWER IMBALANCE protective limits and RCS Variable Low Pressure protective limits for Specification 2.1.1.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(1) DPC-NE-1002A, Reload Design Methodology II, October 1985;

(2) NFS-1001A, Reload Design Methodology, April 1984;

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989;

(4) DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992;

(5) BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November, 1989;

(6) BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, as approved by SER dated February, 1994;

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, August, 1994;

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, February, 1995 and

(9) DPC-NE-3005-P, UFSAR Chapter 15 Transient Analysis Methodology, DPC, JUL97.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

2 5.6.8 Steam Generator Tube Inspection Report

2 The steam generator tube inspection report shall comply with the
2 following:

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes plugged or repaired.

2 4. Number of tubes repaired by rerolling and number of
2 indications detected in the new roll area of the
2 repaired tubes

5.6 Reporting Requirements

2 5.6.8 Steam Generator Tube Inspection Report (continued)

c. Results of steam generator tube inspections which fall into Category C-3 and require notification to the NRC shall be reported prior to resumption of plant operation. The written report shall provide the results of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. The designation of affected and unaffected areas will be reported to the NRC when they are determined.

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(A1) Except as marked 7

Specification 5.0

4.17 STEAM GENERATOR TUBING SURVEILLANCE

Applicability

Applies to the surveillance of tubing of each steam generator.

Objective

To ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

Specification

5.5.10.a ~~4.17.1~~ Examination Methods

Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

5.5.10.b ~~4.17.2~~ Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

For Units 1 and 3, there are a number of steam generator tubes which exceed the tube repair limit as a result of tube end anomalies. These tubes are temporarily exempted from the requirement for sleeving, rerolling or removal from service, until repaired during or before the next Unit 1 and Unit 3 refueling outages (Unit 1 EOC 18, and Unit 3 EOC 17 refueling outages, respectively). An analysis has been performed which confirms that operability of Units 1 and 3 will not be impacted with these tubes in service until the next refueling outage on each of these units.

5.5.10.c ~~4.17.3~~ Selection and Testing (5.5.10-1) (5.5.10.d)

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table ~~4.17-1~~. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in ~~Specification 4.17.4~~ and the inspected tubes shall be verified acceptable per ~~Specification 4.17.5~~. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

5.5.10.e

(1) (a) The first sample inspection during each inservice inspection of each steam generator shall include:

(a) All tubes that previously had detectable wall penetrations (>20%) and have not been plugged or sleeve repaired in the affected area.

(b) At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

(c) A tube adjacent to any selected tube which does not permit passage of the eddy current probe for tube inspection.

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(2) (b.)

Tubes in the following Group(s) may be excluded from the first sample if all tubes in a Group in both OTSG are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

(1) Group A-1: Tubes within one, two, or three rows of the open inspection lane.

(3) (c) All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

5.5.10.1

(4) (d) The tubes selected as the second and third samples (if required by Table ~~4.17-1~~) during each inservice inspection may be subjected to less than a full tube inspection provided:

(a) (a) The tubes selected for these samples include the tubes from those areas of the tubesheet array where tubes with imperfections were previously found.

(b) (a) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but no more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

5.5.10.C.2

(2) Where special inspections are performed pursuant to ~~4.17.2.b~~ defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection, unless the mechanism of degradation is random in nature.

5.5.10.C.2

(3) Where special inspections are performed pursuant to ~~4.17.c~~ defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found in the originally rolled region of the rerolled tubes, need not be included in determining the Inspection Results Category for the general steam generator inspection.

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5.5.10.d

~~4.17.3~~ Inspection Intervals

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies.

1. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of ~~40~~ months.

5.5.10-1

2. If the results of the inservice inspection of a steam generator performed in accordance with Table ~~4.17-1~~ at 40 month intervals fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 months nor more than one fuel cycle after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in ~~4.17A.a~~ and the interval can be extended to a maximum of 40 months.

5.5.10.d.1

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table ~~4.17-1~~ during the shutdown subsequent to any of the following conditions:

5.5.10-1

   1. A seismic occurrence greater than the Operating Basis Earthquake,
   2. A loss-of-coolant accident requiring actuation of the engineered safeguards, or
   3. A main steam line or feedwater line break.

4. After primary to secondary leakage in excess of the limits of Specification ~~3.1.6~~ 3.4.13 an inspection of the affected steam generator will be performed in accordance with the following criteria:

   a. If the leaking tube is in a Group as defined in Section ~~4.17.3.b~~ 5.5.10.c.2 all of the tubes in this Group in this steam generator will be inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

   b. If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the new roll area in the other steam generator.

   c. If the leaking tube is not in a Group as defined in ~~4.17A.d.1~~ 5.5.10.d.4.a then an inspection will be performed on the affected steam generator in accordance with Table ~~4.17-1~~ 5.5.10-1 with an initial inspection sample size of 6% of the tubes in the affected steam generator.

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Definitions

As used in this specification:

Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections.

Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube or a sleeve.

Degraded Tube means a tube or a sleeve containing imperfections ≥20% of the nominal wall thickness caused by degradation.

% Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

Defect means an imperfection of such severity that it exceeds the repair limit. A tube or sleeve containing a defect is defective.

Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

The rerolling repair process will only be used to repair tubes with defects in the upper tubesheet area. The rerolling repair process will be performed only once per steam generator tube using a 1 inch reroll length. The new roll area must be free of degradation in order for the repair to be considered acceptable. The rerolling process used by Oconee is described in the topical report, BAW-2303P, Revision 3.

Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification ~~4.17.4~~ 5.5.10.d

Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

5.6.8 ~~4.17.6~~ ~~Reports~~ Steam Generator Tube Inspection Report

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

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Amendment No. ~~224~~ (Unit 3)

5.6.8

The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes plugged or repaired.

4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the NRC shall be reported pursuant to Specification 6.6.2.1.a prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Bases

The program of periodic inservice inspection of steam generators provides the means to monitor the integrity of the tubing and to maintain surveillance in the event there is evidence of mechanical damage or progressive deterioration due to design, manufacturing errors, or operating conditions. Inservice inspection of the steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures may be taken.

Repair or removal from service will be required for any tube with service-induced metal loss in excess of 40% of the tube or sleeve nominal wall thickness or with a through wall crack. Additional corrective actions may be required to stabilize a circumferentially cracked tube.

The initial sample of tubes inspected in a steam generator includes tubes from three groups. First, lane tubes are inspected to assure their integrity. Second, all other inservice tubes with degradation, inspected in previous inspections, are inspected to assure tube integrity and determine degradation growth, if any. Third, a random sample of 3% of the total number of tubes in both steam generators is inspected. The results of the latter inspection dictate the extent of further examinations.

An objective of this Specification is to provide an inspection plan which will insure, with a high degree of confidence, that no more than 30 defective tubes will remain in a steam generator after an initial C-3 category inspection.

TS 5.5.10-1 Note 2

Following an 18% random inspection (C-3 category inspection) an unaffected area is identified. The unaffected area will be logically and consistently defined based on generator design, defect location and characteristics. The criteria for accepting an area as unaffected depend on the number of defects found in the sample inspected in that area and are established such that there is a 0.05 or smaller probability of accepting the area as unaffected if it contains 30 or more defective tubes.

Experience with Babcock and Wilcox steam generators has indicated that tubes near the open inspection lane are susceptible to forms of degradation unique to that area. Therefore, tubes within one, two, or three rows of the inspection lane have been defined as a special group. If all of these tubes are inspected

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5.5.10 Steam Generator (SG) Tube Surveillance Program

2 This program provides the controls for SG tube surveillance. The
2 program shall include the following:

a. Examination Methods

Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

b. Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

2 For Units 1 and 3, there are a number of steam generator
2 tubes which exceed the tube repair limit as a result of tube
2 end anomalies. These tubes are temporarily exempted from
2 the requirements for sleeving, rerolling or removal from
2 service, until repaired during or before the next Unit 1 and
2 Unit 3 refueling outages (Unit 1 EOC 18, Unit 3 EOC 17
2 refueling outages, respectively). An analysis has been
2 performed which confirms the operability of Units 1 and 3
2 will not be impacted with these tubes in service until the
2 next refueling outage on each of these units.

c. Selection and Testing

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.10-1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in 5.5.10.d and the inspected tubes shall be verified acceptable per 5.5.10.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

1. The first sample inspection during each inservice inspection of each steam generator shall include:

a. All tubes that previously had detectable wall penetrations (>20%) and have not been plugged or sleeve repaired in the affected area.

b. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

c. A tube adjacent to any selected tube which does not permit passage of the eddy-current probe for tube inspection.

2. Tubes in the following Group(s) may be excluded from the first sample if all tubes in a Group in both OTSG are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

Group A-1: Tubes within one, two, or three rows of the open inspection lane.

2 3. All tubes which have been repaired using the reroll
2 process will have the new roll area inspected during
2 the inservice inspection.
2
2 4. The tubes selected as the second and third samples (if required by Table 5.5.10-1) during each inservice inspection may be subjected to less than a full tube inspection provided:

a. The tubes selected for these samples include the tubes from those areas of the tubesheet array where tubes with imperfections were previously found.

b. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but no more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are

degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 5.5.10.c.2, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection, unless the mechanism of degradation is random in nature.

2 (3) Where special inspections are performed pursuant
2 to 5.5.10.c.2, defective or degraded tube
2 indications found in the new roll area as a
2 result of the inspection and any indications
2 found in the originally rolled region of the
2 rerolled tube, need not be included in
2 determining the Inspection Results Category for
2 the general steam generator inspection.

d. Inspection Intervals

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies.

1. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections following service under all volatile treatment (AVT) conditions fall into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of 40 months.

2. If the results of the inservice inspection of a steam generator performed in accordance with Table 5.5.10-1 at 40 month intervals fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 months nor more than one fuel cycle after the previous inspection. The increase in inspection frequency shall apply until a subsequent

inspection meets the conditions specified in 5.5.10.d.1 and the interval can be extended to a maximum of 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.10-1 during the shutdown subsequent to any of the following conditions:

   a. A seismic occurrence greater than the Operating Basis Earthquake,

   b. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   c. A main steam line or feedwater line break.

4. After primary to secondary leakage in excess of the limits of Specification 3.4.13, an inspection of the affected steam generator will be performed in accordance with the following criteria:

   a. If the leaking tube is in a Group as defined in Section 5.5.10.c.2, all of the tubes in this Group in this steam generator will be inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

2 b. If the leaking tube has been repaired by the reroll
2 process and is leaking in the new roll area, all tubes in
2 the steam generator that have been repaired by the reroll
2 process will have the new roll area inspected. If he
2 results of this inspection fall into the C-3 category,
2 additional inspections will be performed in the new roll
2 area in the other steam generator.
2
2 c. If the leaking tube is not in a Group as defined in 5.5.10.d.4.a, then an inspection will be performed on the affected steam generator in accordance with Table 5.5.10-1 with an initial inspection sample size of 6% of the tubes in the affected steam generator.

e. Definitions

As used in this specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if

detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube or a sleeve.

3. Degraded Tube means a tube or a sleeve containing imperfections $\geq$ 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube or sleeve containing a defect is defective.

6. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness.

   The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

2 The rerolling repair process will only be used to repair
2 tubes with defects in the upper tubesheet area. The
2 rerolling repair process will be performed only once per
2 steam generator tube using a 1 inch reroll length. The new
2 roll area must be free of degradation in order for the
2 repair to be considered acceptable. The rerolling process
2 used by Oconee is described in the Topical Report, BAW-2303,
2 Revision 3.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.10.d.
8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

TABLE 5.5.10-1 (page 1 of 2)
STEAM GENERATOR TUBE INSPECTION

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this SG.	C-1	None
					C-2	Plug or repair defective tubes.
					C-3	Plug or repair defective tubes and perform action for C-3 result of 1st Sample.
			C-3	Plug or repair defective tubes and perform actions for C-3 results on 1st Sample.	N/A	N/A

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
(continued)	C-3	Inspect 6S tubes in the S.G, plug or repair defective tubes and inspect 2S tubes in the other S.G. Perform follow-on inspections in the other S.G. in accordance with results of the above inspection as applied to Table 5.5.10-1 Prompt notification to NRC pursuant to 10 CFR 50.72	C-1	N/A	N/A	N/A
			C-2	N/A	N/A	N/A
			C-3 (2)	(a) If defects can be localized to an affected area, inspect all tubes in affected area and plug or repair defective tubes. (b) If defects cannot be localized to an affected area, inspect all tubes in this S.G. and plug or repair defective tubes.	C-1 C-2 C-3	N/A N/A N/A

Notes: (1) S=3(N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

(2) Following an 18% random inspection (C-3 category inspection) an unaffected area is identified. The unaffected area will be logically and consistently defined based on generator design, defect location and characteristics. The criteria for accepting an area as unaffected depend on the number of defects found in the sample inspected in that area and are established such that there is a 0.05 or smaller probability of accepting the area as unaffected if it contains 30 or more defective tubes.