Southern Nuclear
Operating Company, Inc.
Post Office Drawer 470
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SOUTHERN COMPANY
Energy to Serve Your World SM

FNP-023-NRC-DC

MAY 15, 2000

REGION II NRC
61 FORSYTH CENTER SUITE 23T85
ATLANTA, GA. 30303

DEAR SIR,

ATTACHED IS AMENDMENT NO. 147 TO THE TECHNICAL SPECIFICATIONS. PLEASE FILE IN THE NEW IMPROVED TECH SPEC BOOK.

IF YOU HAVE QUESTIONS PLEASE CALL ME AT 334-899-5156 EXTENSION 3402.

SINCERELY,

Donnie Hardy
DONNIE HARDY LB

DOCUMENT CONTROL SUPERVISOR

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Memo Disk 1 - NRC

ATTACHMENT TO LICENSE AMENDMENT No. 147

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of Facility Operating License No. NPF-2 with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating area of changes. Pages noted with an "*" have changed only due to information rolling over from one page to another.

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Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
11. Reactor Coolant Pump (RCP) Breaker Position						
a. Single Loop	1(g)	1 per RCP	N	SR 3.3.1.12	NA	NA
b. Two Loops	1(h)	1 per RCP	M	SR 3.3.1.12	NA	NA
12. Undervoltage RCPs	1(f)	3	M	SR 3.3.1.6 SR 3.3.1.10	≥ 2640 V	≥ 2680 V
13. Underfrequency RCPs	1(f)	3	M	SR 3.3.1.6 SR 3.3.1.10	≥ 56.9 Hz	≥ 57 Hz
14. Steam Generator (SG) Water Level — Low Low	1,2	3 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	≥ 27.6% (k) ≥ 24.6% (l)	≥ 28% (k) ≥ 25% (l)

(f) Above the P-7 (Low Power Reactor Trips Block) interlock.

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

(k) Unit 1 only (after Steam Generator Replacement)

(l) Unit 2 only (before Steam Generator Replacement)

Table 3.3.1-1 (page 5 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
15. Turbine Trip						
a. Low Auto Stop Oil Pressure	1 (i)	3	O	SR 3.3.1.10 SR 3.3.1.13	≥ 43 psig	≥ 45 psig
b. Turbine Throttle Valve Closure	1 (i)	4	P	SR 3.3.1.10 SR 3.3.1.13	NA	NA
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.12	NA	NA
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2 (d)	2	S	SR 3.3.1.10 SR 3.3.1.11	≥ 6E-11 amp	≥ 1E-10 amp
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	NA	NA	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.10 SR 3.3.1.11	≤ 30.4% RTP	≤ 30% RTP
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.10 SR 3.3.1.11	≤ 50.4% RTP	≤ 50% RTP
e. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.10 SR 3.3.1.11	≥ 7.6% RTP and ≤ 10.4% RTP	≥ 8% RTP and ≤ 10% RTP
f. Turbine Impulse Pressure, P-13	1	2	T	SR 3.3.1.1 SR 3.3.1.10 SR 3.3.1.11	≤ 11% turbine power	≤ 10% turbine power

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(i) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.2-1 (page 4 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2	2 trains	H	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8	NA	NA
b. SG Water Level - High High (P-14)	1,2	3 per SG	I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≤ 82.4% (h) ≤ 78.9% (i)	≤ 82% (h) ≤ 78.5% (i)
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8	NA	NA
b. SG Water Level - Low Low	1,2,3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9(g)	≥ 27.6% (h) ≥ 24.6% (i)	≥ 28% (h) ≥ 25% (i)
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. Undervoltage Reactor Coolant Pump	1,2	3	I	SR 3.3.2.5 SR 3.3.2.7 SR 3.3.2.9	≥ 2640 volts	≥ 2680 volts
e. Trip of all Main Feedwater Pumps	1	2 per pump	J	SR 3.3.2.10	NA	NA
7. ESFAS Interlocks						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	L	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8	NA	NA
b. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	C	SR 3.3.2.6	NA	NA
c. Pressurizer Pressure, P-11	1,2,3	3	K	SR 3.3.2.4 SR 3.3.2.7	≤ 2003 psig	≤ 2000 psig
d. T_{avg} - Low Low, P-12 (Decreasing) (Increasing)	1,2,3	1 per loop	K	SR 3.3.2.4 SR 3.3.2.7	≥ 542.6°F ≤ 545.4°F	≥ 543°F ≤ 545°F

(g) Applicable to MDAFW pumps only.
(h) Unit 1 only (after Steam Generator Replacement)
(i) Unit 2 only (before Steam Generator Replacement)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops—MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and either:

a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

--NOTE--
All reactor coolant pumps may not be in operation for ≤ 1 hour per 8 hour period provided:

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

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

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required RCS loop inoperable.	A.1	Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 4.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal.	C.1 Restore required RCS loop to operation. OR C.2 De-energize all control rod drive mechanisms (CRDMs).	1 hour 1 hour
D. Two required RCS loops inoperable. OR No RCS loop in operation.	D.1 De-energize all CRDMs. AND D.2 Suspend all operations involving a reduction of RCS boron concentration. AND D.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	12 hours
SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ [Unit 1 only: 30%] [Unit 2 only: 28%] (narrow range) for required RCS loops.	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

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 residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

---NOTES---

1. All reactor coolant pumps (RCPs) and RHR pumps may not be in operation for ≤ 2 hours per 8 hour period provided:

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

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

2. No RCP shall be started with any RCS cold leg temperature ≤ 325°F unless:

    a. The secondary side water temperature of each steam generator (SG) is < 50°F above each of the RCS cold leg temperatures; or

    b. The pressurizer water volume is less than 770 cubic feet (24% of wide range, cold, pressurizer level indication).

APPLICABILITY: MODE 4.

ACTIONS

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required RHR loop inoperable. AND Two required RCS loops inoperable.	B.1 Be in MODE 5.	24 hours
C. Required RCS or RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1 Suspend all operations involving a reduction of RCS boron concentration. AND C.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side water levels are ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range) for required RCS loops.	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops — MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least two steam generators (SGs) shall be ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range).

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

1. The RHR pump of the loop in operation may not be in operation for ≤ 2 hours per 8 hour period provided:

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

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

2. One required RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ 325°F unless:

   a. The secondary side water temperature of each SG is < 50°F above each of the RCS cold leg temperatures; or

   b. The pressurizer water volume is less than 770 cubic feet (24% of wide range, cold, pressurizer level indication).

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

5. The number of operating Reactor Coolant Pumps is limited to one at RCS temperatures < 110°F with the exception that a second pump may be started for the purpose of maintaining continuous flow while taking the operating pump out of service.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. <u>AND</u> Required SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. <u>OR</u> A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately Immediately
B. Required RHR loops inoperable. <u>OR</u> No RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration. <u>AND</u> B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	12 hours
SR 3.4.7.2	Verify SG secondary side water level is ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range) in required SGs.	12 hours

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. 450 gallons per day total primary to secondary LEAKAGE through all steam generators (SGs); and

e. 150 gallons 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. <u>OR</u> Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

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. --- Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	-------NOTE-------- Only required to be performed during steady state operation ---------------------- 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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. DOSE EQUIVALENT I-131 [Unit 1 only: > 0.5 μCi/gm] [Unit 2 only: > 0.30 μCi/gm].	--------------------Note-------------------- LCO 3.0.4 is not applicable. ---		
	A.1	Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	AND		
	A.2	Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1	Be in MODE 3 with T_{avg} < 500°F.	6 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1	Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/$\bar{E}$ μCi/gm.	7 days
SR 3.4.16.2	-----------------------------NOTE----------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity [Unit 1 only: ≤ 0.5 μCi/gm] [Unit 2 only: ≤ 0.30 μCi/gm].	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

SURVIELLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	-------------------------------NOTE---------------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ---	
	Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

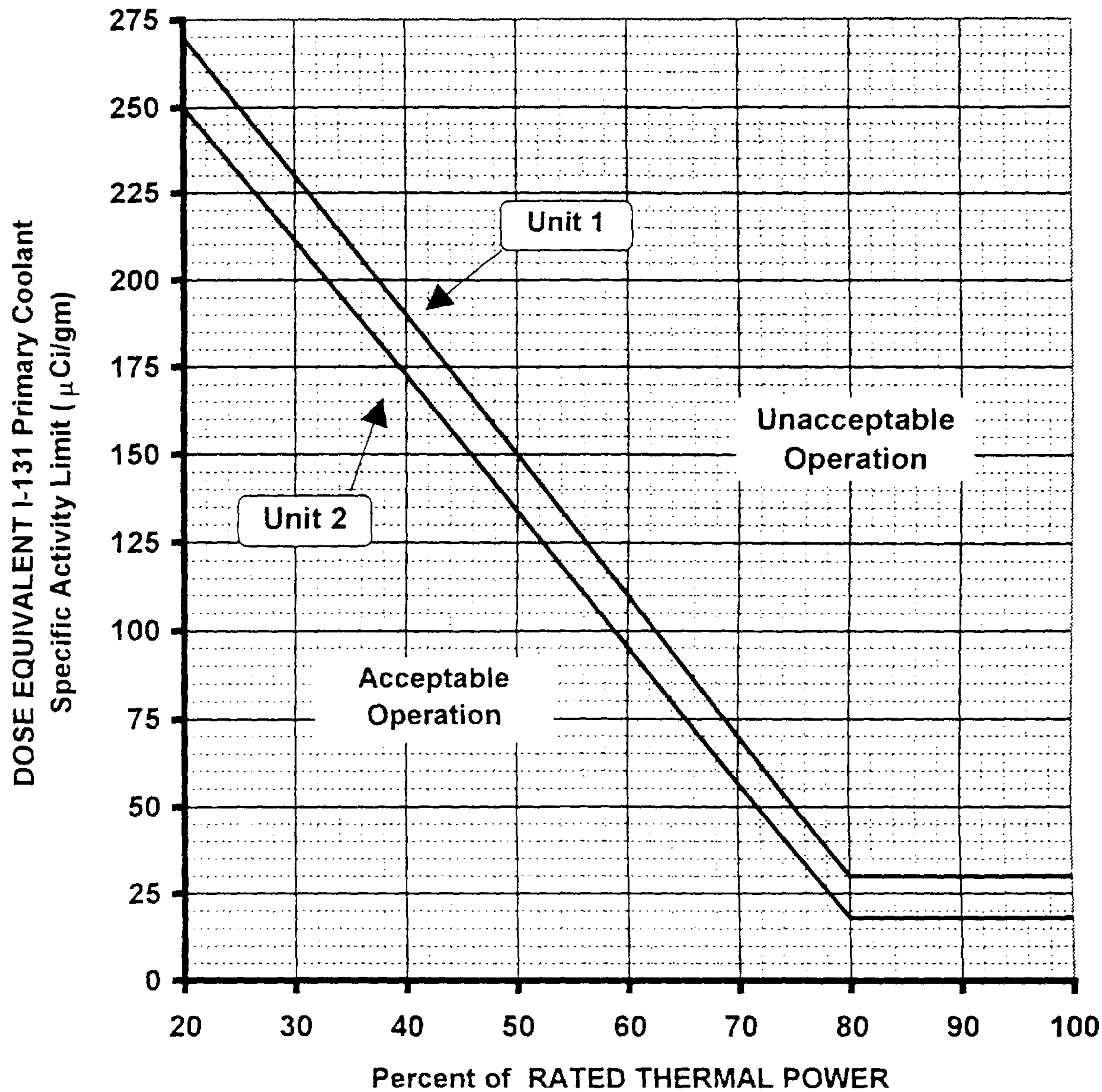

Figure 3.4.16-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity [Unit 1 only: > 0.5 μCi/gm] [Unit 2 only: > 0.30 μCi/gm] DOSE EQUIVALENT I-131.

5.5 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

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

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program Test Frequencies. [Specification 5.5.9 is not required to be performed on the replacement steam generators during the shutdown when the steam generators are replaced.]

5.5.9.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

5.5.9.1 Steam Generator Sample Selection and Inspection

Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5.9-1.

5.5.9.2 Steam Generator Tube# Sample Selection and Inspection

5.5.9.2.1 The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables 5.5.9-2 and 5.5.9-3. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.9.3, and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators. [Unit 2 only: Selection of tubes to be inspected is not affected by the F* designation. When applying the exceptions of 5.5.9.2.1.a through 5.5.9.2.1.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection.] The tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%.

[Unit 2 only] When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 5.5.9.4.a.9.

(continued)

5.5.9.2.1 (continued)

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 5.5.9.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube [Unit 2 only: or sleeve] inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

4. [Unit 2 only] Indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

c. The tubes selected as the second and third samples (if required by Tables 5.5.9-2 and 5.5.9-3) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

(continued)

5.5.9.2.1 (continued)

d. [Unit 2 only] Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

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 not 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.

Note: In all inspections, previously degraded tubes [Unit 2 only: or sleeves] must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

5.5.9.2.2 [Unit 2 only] Steam Generator F* Tube Inspection

In addition to the minimum sample size as determined by Specification 5.5.9.2.1, all F* tubes will be inspected within the tubesheet region. The results of this inspection will not be a cause for additional inspections per Tables 5.5.9-2 and 5.5.9-3.

5.5.9.3 Inspection Frequencies

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling 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 once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Tables 5.5.9-2 and 5.5.9-3 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 5.5.9-2 and 5.5.9-3 during the shutdown subsequent to any of the following conditions:

(continued)

5.5 Programs and Manuals

5.5.9.3 Inspection Frequencies (continued)

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of Specification 3.4.13.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

5.5.9.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube [Unit 2 only: or sleeve] from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube [Unit 2 only: or sleeve].

3. Degraded Tube means a tube [Unit 2 only:, including the sleeve if the tube has been repaired,] that contains imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube [Unit 2 only: or sleeve] wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging [Unit 2 only: or repair limit]. A tube [Unit 2 only: or sleeve] containing a defect is defective.

(continued)

5.5 Programs and Manuals

5.5.9.4 Acceptance Criteria (continued)

6. [Unit 1 only] Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness.

   [Unit 2 only] Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be repaired (i.e., sleeved) or removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness. This definition does not apply for tubes that meet the F* criteria. For a tube that has been sleeved with a mechanical joint sleeve, through wall penetration of greater than or equal to 31% of sleeve nominal wall thickness in the sleeve requires the tube to be removed from service by plugging. For a tube that has been sleeved with a welded joint sleeve, through wall penetration greater than or equal to 24% of sleeve nominal wall thickness in the sleeve between the weld joints requires the tube to be removed from service by plugging. This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 5.5.9.4.a.14 for the repair limit applicable to these intersections. For a tube with an imperfection or flaw in the tube sheet below the lower joint of an installed elevated laser welded sleeve, no repair or plugging is required provided the installed sleeve meets all sleeved tube inspection requirements.

7. Unserviceable describes the condition of a tube [Unit 2 only: or sleeve] 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 5.5.9.3.c, above.

(continued)

5.5 Programs and Manuals

5.5.9.4 Acceptance Criteria (continued)

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. [Unit 2 only: For a tube with a tube sheet sleeve installed, the point of entry is the bottom of the tube sheet sleeve below the lower sleeve joint. For a tube that has been repaired by sleeving, the tube inspection should include the sleeved portion of the tube.]

9. [Unit 2 only] Tube repair refers to mechanical sleeving, as described by Westinghouse report WCAP-11178, Rev. 1, or laser welded sleeving, as described by Westinghouse reports WCAP-13088, Revision 4, and WCAP-14740 dated January 1997, which is used to maintain a tube in service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure.

10. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed using the equipment and techniques expected to be used during subsequent inservice inspections.

11. [Unit 2 only] F* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.60 inches plus allowance for eddy current uncertainty measurement and is measured down from the top of the tube sheet or the bottom of the roll transition, whichever is lower in elevation. The allowance for eddy current uncertainty is documented in the steam generator eddy current inspection procedure.

(continued)

5.5.9.4 Acceptance Criteria (continued)

12. [Unit 2 only] F* Tube is a tube:

    a. with degradation equal to or greater than 40% below the F* distance, and

    b. which has no indication of imperfections greater than or equal to 20% of nominal wall thickness within the F* distance, and

    c. that remains inservice.

13. [Unit 2 only] Tube Expansion is that portion of a tube which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the tube and the hole in the tubesheet. Tube expansion also refers to that portion of a sleeve which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the sleeve and the parent steam generator tube.

14. [Unit 2 only] Tube Support Plate Repair Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:

    a. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to the lower voltage repair limit (2.0 volts), will be allowed to remain in service.

    b. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (2.0 volts), will be repaired or plugged except as noted in 5.5.9.4.a.14.c below.

(continued)

5.5 Programs and Manuals

5.5.9.4 Acceptance Criteria (continued)

c. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (2.0 volts), but less than or equal to the upper voltage repair limit*, may remain in service if a rotating probe inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit*, will be plugged or repaired.

d. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9.4.a.14.a, 5.5.9.4.a.14.b, and 5.5.9.4.a.14.c.

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\frac{[CL - \Delta t]}{CL}}$$

$$V_{MLRL} = V_{MURL} - [V_{URL} - V_{LRL}]\frac{[CL - \Delta t]}{CL}$$

where:

V_{URL} = upper voltage repair limit
V_{LRL} = lower voltage repair limit
V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

* The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

(continued)

5.5.9.4 Acceptance Criteria (continued)

Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 5.5.9.4.a.14.a, 5.5.9.4.a.14.b, and 5.5.9.4.a.14.c.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging [Unit 2 only: or repair] of all tubes exceeding the plugging [Unit 2 only: or repair] limit) required by Tables 5.5.9-2 and 5.5.9-3.

Table 5.5.9-1

No. of Steam Generators per Unit	Three
First Inservice Inspection	Two
Second and Subsequent Inservice Inspections	One*

* The other steam generator not inspected during the first inservice inspection shall be reinspected. The third and subsequent inspections may be limited to one steam generator on a rotating schedule encompassing 3 N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the same sequence shall be modified to inspect the most severe conditions.

Table 5.5.9-2
Steam Generator Tube Inspection

<table>
<tr><th></th><th colspan="2">1st Sample Inspection</th><th colspan="2">2nd Sample Inspection</th><th colspan="2">3rd Sample Inspection</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per SG</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C-2</td><td rowspan="5">Plug [Unit 2 only: or repair] defective tubes and inspect additional 2S tubes in this SG</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-2</td><td rowspan="3">Plug [Unit 2 only: or repair] defective tubes and inspect additional 4S tubes in this SG</td><td>C-1</td><td>None</td></tr>
<tr><td>C-2</td><td>Plug [Unit 2 only: or repair] defective tubes</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect all tubes in this SG, plug [Unit 2 only: or repair] defective tubes and inspect 2S tubes in each other SG

Notification to NRC pursuant to 10 CFR 50.73</td><td>All other SGs are C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Some SGs C-2 but no additional SGs are C-3</td><td>Perform action for C-2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Additional SG is C-3</td><td>Inspect all tubes in each SG and plug [Unit 2 only: or repair] defective tubes. Notification to NRC pursuant to 10 CFR 50.73</td><td>N/A</td><td>N/A</td></tr>
</table>

$S = \frac{3N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

[Unit 2 only — NOTE: F* tubes do not have to be plugged or repaired.]

Table 5.5.9-3
Steam Generator Repaired Tube Inspection
[Unit 2 only]

	1st Sample Inspection		2nd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes (1)(2)	C-1	None	N/A	N/A
	C-2	Plug or repair defective repaired tubes and inspect 100% of the repaired tubes in this SG	C-1	None
			C-2	Plug or repair defective repaired tubes
			C-3	Perform action for C-3 result of first sample.
	C-3	Inspect all repaired tubes in this SG, plug or repair defective tubes and inspect 20% of the repaired tubes in each SG Notification to NRC pursuant to 10 CFR 50.72(b)(2).	All other SGs are C-1.	None
			Some SGs C-2 but no additional SGs are C-3.	Perform action for C-2 result of first sample.
			Additional SG is C-3.	Inspect all repaired tubes in each SG and plug or repair defective tubes. Notification to NRC pursuant to 10 CFR 50.72(b)(2).

(1) Each repair method is considered a separate population for determination of scope expansion

(2) The inspection of repaired tubes may be performed on tubes from 1 to 3 steam generators based on outage plans

5.5 Programs and Manuals

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b 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).

5.5.15 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 actions may 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;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of 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

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

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.

5.5.16 Main Steamline Inspection Program

The three main steamlines from the rigid anchor points of the containment penetrations downstream to and including the main steam header shall be inspected. The extent of the inservice examinations completed during each inspection interval (IWA 2400, ASME Code, 1974 Edition, Section XI) shall provide 100 percent volumetric examination of circumferential and longitudinal pipe welds to the extent practical. The areas subject to examination are those defined in accordance with examination category C-G for Class 2 piping welds in Table IWC-2520.

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is [Unit 1 only: 43.8 psig] [Unit 2 only: 43 psig].

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

(continued)

5.6 Reporting Requirements

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "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.

5.6.9 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.

5.6.10 Steam Generator Tube Inspection Report

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged [Unit 2 only: , repaired or designated F*,], in each steam generator shall be reported to the Commission within 15 days of the completion of the plugging [Unit 2 only: or repair] effort.

b. The complete results of the steam generator tube [Unit 2 only: and sleeve] inservice inspection shall be submitted to the Commission within 12 months following the completion of the inspection. This Report shall include:

   1. Number and extent of tubes [Unit 2 only: and sleeves] inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged [Unit 2 only: or repaired].

(continued)

5.6 Reporting Requirements

5.6.10 Steam Generator Tube Inspection Report (continued)

c. Results of steam generator tube inspections which fall into Category C-3 shall be considered a Reportable Event and shall be reported pursuant to 10 CFR 50.73 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

d. [Unit 2 only] For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service (Mode 4) should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (detemined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

Replace the following pages of the ITS Bases with the attached revised pages. The revised pages are identified as Revision 1 and contain vertical lines indicating area of changes (except LOEP has no rev bars). Pages noted with an "*" have changed only due to information rolling over from one page to another.

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BASES

ACTIONS

C.1 and C.2 (continued)

inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the RTBs must be opened.

The Completion Times of 1 hour to restore the required RCS loop to operation or de-energize all CRDMs is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

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

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ [Unit 1 only: 30%] [Unit 2 only: 28%] for required RCS loops. If the SG secondary side narrow range

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.2 (continued)

water level is < [Unit 1 only: 30%] [Unit 2 only: 28%], the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

REFERENCES

None.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side wide range water level is ≥ [Unit 1 only: 75%] [Unit 2 only: 74%]. If the SG secondary side wide range water level is < [Unit 1 only: 75%] [Unit 2 only: 74%], the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

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

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops — MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining two SGs with secondary side water levels ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range) to provide an alternate method for decay heat removal via natural circulation (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops — MODE 5 (Loops Filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range). One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range). Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all RHR pumps to not be in operation ≤ 2 hours per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 2 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration

(continued)

BASES

LCO (continued)

distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be < 50°F above each of the RCS cold leg temperatures or that the pressurizer water volume is less than 770 cubic feet (24% of wide range, cold, pressurizer level indication) before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ 325°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

Note 5 restricts the number of operating reactor coolant pumps at RCS temperatures less than 110°F. Only one reactor coolant pump is allowed to be in operation below 110°F (except during pump swap operations) consistent with the assumptions of the P/T Limits Curve.

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

BASES

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range).

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops — MODES 1 and 2";
LCO 3.4.5, "RCS Loops — MODE 3";
LCO 3.4.6, "RCS Loops — MODE 4";
LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels < [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range), redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal.

The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side wide range water levels are ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side water level is ≥ [Unit 1 only: 75%] [Unit 2 only: 74%] (wide range) in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

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

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is typically seen as a precursor to a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 150 gpd per SG primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via the main steam safety valves. The majority of the activity released to the atmosphere results from the tube rupture. Therefore, the 150 gpd per SG primary to secondary LEAKAGE is inconsequential.

[Unit 1 Only] The SLB is more limiting for primary to secondary LEAKAGE. The safety analysis for the SLB assumes 500 gpd and 470 gpd primary to secondary LEAKAGE in the ruptured and intact steam generators respectively as an initial condition. The dose consequences resulting from the SLB accident are bounded by a small fraction (i.e., 10%) of the limits defined in 10 CFR 100. The RCS specific activity assumed was a bounding value of 1.0 μCi/gm DOSE EQUIVALENT I-131, with either a pre-existing or an accident initiated iodine spike. These values bound the Technical Specifications values.

[Unit 2 Only] The SLB is more limiting for primary to secondary LEAKAGE. The safety analysis for the SLB assumes 500 gpd primary to secondary LEAKAGE in one steam generator as an initial condition. The Unit 2 MSLB analysis in support of Generic Letter 95-05 has shown that steam generator tube leakage of 11.8 gpm in the faulted loop, and 0.1 gpm (approximately 150 gpd) in each of the intact loops (total leakage of 12 gpm), following a SLB outside of containment, but upstream of the main steam isolation valves, results in offsite doses bounded by a small fraction (i.e., 10%) of the 10 CFR 100 guidelines. The RCS specific activity assumed was 0.30 μCi/gm DOSE EQUIVALENT I-131, with either a pre-existing or an accident initiated iodine spike. These values bound the Technical Specifications values.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

The limits for total primary to secondary LEAKAGE through all SGs produce acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

(continued)

BASES

LCO (continued)

e. Primary to Secondary LEAKAGE through Any One SG

The limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident, or for the duration of the accident at the Low Population Zone, is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to an appropriate fraction of the 10 CFR 100 limits (i.e., a small fraction of or well within the 10 CFR 100 limits depending on the specific accident analysis) during analyzed transients and accidents.

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

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to an appropriate fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR or MSLB accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting doses will not exceed an appropriate fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at [Unit 1 only: 1.0 μCi/gm] [Unit 2 only: 0.5 μCi/gm] and a bounding reactor coolant steam generator (SG) tube leakage of [Unit 1 only: 1 gpm total for three SGs] [Unit 2 only: 150 gpd per SG]. The MSLB analysis assumes a steam generator tube leakage of [Unit 1 only: 500 gpd] [Unit 2 only 11.8 gpm] in the faulted loop and [Unit 1 only: 470 gpd] [Unit 2 only: 150 gpd] in each of the intact loops for a total leakage of [Unit 1 only: 1440 gpd] [Unit 2 only: 12 gpm].

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

This analysis resulted in offsite doses bounded by a small fraction (i.e., 10%) of the 10 CFR 100 guidelines using ICRP Dose Conversion Factors (DCFs). The initial RCS specific activity assumed was [Unit 1 only: 1.0 μCi/gm] [Unit 2 only: 0.30 μCi/gm] DOSE EQUIVALENT I-131 with an iodine spike. These values bound the Technical Specifications values. The safety analysis assumes for both the SGTR and MSLB the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.16, "Secondary Specific Activity."

The analysis for the MSLB accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The SGTR analysis assumes an RCS coolant activity of [Unit 1 only: 1.0 μCi/gm] [Unit 2 only: 0.5 μCi/gm] DOSE EQUIVALENT I-131. The MSLB analysis considers two cases of reactor coolant specific activity. One case assumes specific activity at [Unit 1 only: 1.0 μCi/gm] [Unit 2 only: 0.30 μCi/gm] DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity release rate into the reactor coolant by a factor of 500 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at [Unit 1 only: 60 μCi/gm] [Unit 2 only: 18 μCi/gm] DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. These values bound the Technical Specifications values. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The SGTR analysis also assumes a loss of offsite power coincident with a reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

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

The MSLB analysis assumes a double-ended guillotine break of a main

BASES

APPLICABLE SAFETY ANALYSES (continued)

rapidly depressurize and release both the radionuclides initially contained in the secondary coolant, and the primary coolant activity transferred via SG tube leakage, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact SGs and noble gas activity due to SG tube leakage is released to the atmosphere through either the SG atmospheric relief valves (ARVs) or the SG safety relief valves.

The safety analysis assumes an accident initiated iodine spike and shows the radiological consequences of a MSLB accident are within a small fraction of the Reference 1 dose guideline limits.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The MSLB safety analysis has concurrent and pre-accident iodine spiking levels up to [Unit 1 only: 60 μCi/gm] [Unit 2 only: 18 μCi/gm] DOSE EQUIVALENT I-131.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a MSLB accident occurring during the established 48 hour time limit. The occurrence of a MSLB accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

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

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 0.5 μCi/gm DOSE EQUIVALENT I-131 for the SGTR analysis and [Unit 1 only: 0.5 μCi/gm] [Unit 2 only: 0.30 μCi/gm] DOSE EQUIVALENT I-131 for the MSLB analysis, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the thyroid dose to an individual during the Design Basis Accident (DBA) will be an appropriate fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

(continued)

BASES

LCO (continued)

The SGTR (Ref. 2) and MSLB accident analyses show that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR or MSLB, lead to site boundary doses that exceed the dose guideline limits.

APPLICABILITY

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

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

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

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

A Note to the Required Action of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a three way post tensioning system.

The concrete reactor building is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

    1. capable of being closed by an OPERABLE automatic containment isolation system, or

(continued)

BASES

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. All equipment hatches are closed; and

d. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.15% of containment air weight per day for the first 24 hours and 0.075% thereafter (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.15% per day in the safety analysis at P_a = [Unit 1 only: 43.8 psig] [Unit 2 only: 43 psig] (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

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

The personnel air lock is nominally a right circular cylinder, 10 ft in diameter, with a door at each end. The auxiliary hatch is nominally a right circular cylinder, 6 ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches and mechanical pointers for both doors that provide local indication of door position. With power supplied to the door operators, this indication is provided by position indication lights. With power removed from the door operators, this indication is provided by mechanical pointers located beside each door's manual handwheels. A set of handwheels, indicating lights, and manual pointers is located inside the air locks, and on the outside of the air locks on both the auxiliary building and containment sides.

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

BASES

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident, a rod ejection accident, and a fuel handling accident in containment (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.15% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as L_a, the maximum allowable containment leakage rate at the calculated peak containment internal pressure, P_a ([Unit 1 only: 43.8 psig] [Unit 2 only: 43 psig]), following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). An inadvertent actuation of the Containment Spray System is not part of the containment pressure response licensing basis for Farley.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case SLB generates larger mass and energy release than the worst case LOCA. Thus, the SLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 17.7 psia (3.0 psig). This resulted in a maximum peak pressure from a SLB of [Unit 1 only: 52.0 psig] [Unit 2 only: 52.4 psig]. The containment analysis (Ref. 1) shows the maximum peak calculated containment pressure, P_a, resulting from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, [Unit 1 only: 43.8 psig] [Unit 2 only: 43.0 psig], does not exceed the containment design pressure, 54 psig.

The containment was also designed for an external pressure load equivalent to –3.0 psig to account for the external loading from tornado depressurization.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2). Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure due to tornado induced atmospheric depressurization.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3, and 4. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System, and Containment Cooling System being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is a SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 127°F. This resulted in a maximum containment air temperature of [Unit 1 only: 367°F] [Unit 2 only: 383°F]. The design air temperature is 378°F.

[Unit 1 only] The temperature limit is used to establish the environmental qualification operating envelope for containment. The basis of the containment design air temperature is to ensure the performance of safety-related equipment inside containment (Ref. 2). Thermal analyses show that the containment air temperature remains below the equipment design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA SLB.

[Unit 2 only] The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design air temperature for only a few seconds during the transient. The basis of the containment design air temperature, however, is to ensure the performance of safety-related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design air temperature was short enough that the equipment surface temperatures remained below the equipment design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA SLB.

The temperature limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a SLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant containment structure peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

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

B.1 and B.2

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurements taken at four of the following sensor locations with at least two being containment air cooler intake sensors:

Instrument Number	Sensor Location
TE3187 E, F, G, & H	Containment Air Cooler Intake
TE3188 H & I	Lower Compartment
TE3188 J	Reactor (lower)

The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. FSAR, Section 6.2.
2. 10 CFR 50.49.

BASES

BACKGROUND

Containment Cooling System (continued)

ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post accident operation following an actuation signal, unless an LOSP signal is present, the Containment Cooling System fans are designed to start automatically in slow speed if not already running. If an LOSP signal is present, only the two fans selected (one per train) will receive an auto-start signal and will start in slow speed. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere. In addition, if temperature at the cooler discharge reaches 135°F, fusible links holding dropout plates will open and the fan discharge will no longer be directed through the common discharge header. This function helps to protect the fans in a post-accident environment by reducing the back pressure on the fans. The temperature of the SW is an important factor in the heat removal capability of the fan units.

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure and results in one train of the Containment Spray System and Containment Cooling System being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is [Unit 1 only: 52.0 psig] [Unit 2 only: 52.4 psig] (experienced during a SLB). The analysis shows that the peak containment temperature is [Unit 1 only: 367°F] [Unit 2 only: 383°F] (experienced during a SLB). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5, "Containment Air Temperature," for a detailed discussion).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analyses and evaluations assume a unit specific power level of 102%, one containment spray train and one containment cooling fan operating, and initial (pre-accident) containment conditions of 127°F and -1.0 to +3.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

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

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -2.9 psig containment pressure and is associated with the sudden cooling effect in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-3 pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time of 56 seconds includes diesel generator (DG) startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 4).

Containment cooling train performance for post accident conditions is given in Reference 3. The result of the analysis is that each train having at least one OPERABLE fan unit with at least 600 gpm SW flow can provide 100% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 5.

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-1 pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow.

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B 3.7 PLANT SYSTEMS

B 3.7.16 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 450 gallons per day tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of [Unit 1 only: 0.5 μCi/gm] [Unit 2 only: 0.30 μCi/gm] (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the site boundary would be within the limits of 10 CFR 20.1001- 20.2402 if the main steam safety valves (MSSVs) and Atmospheric Relief Valves (ARVs) are open for 2 hours following a trip from full power.

Operating at the allowable limits results in a 2 hour site boundary exposure well within the 10 CFR 100 (Ref. 1) limits.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the site boundary limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric relief valves (ARVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ARVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant in the steam generators ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.