NMC
Committed to Nuclear Excellence
Nuclear Management Company, LLC

L-HU-06-001
10 CFR 50.90

February 16, 2006

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Point Beach Nuclear Plant Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), the Nuclear Management Company, LLC (NMC) is submitting a request for an amendment to the technical specifications (TS) for the above identified facilities.

The proposed amendment would revise the TS requirements related to steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

Enclosure 1 provides a description of the proposed change and confirmation of applicability. Enclosures 2A, 2B and 2C provide plant specific clarifications of TSTF-449 with respect to each facility's TS and Bases. Enclosures 3A, 3B and 3C provide unit specific steam generator information. Enclosures 4A, 4B and 4C provide the existing TS and Bases pages marked-up to show the proposed change. Enclosures 5A, 5B and 5C provide the revised TS pages.

NMC requests approval of the proposed License Amendment within one year of the submittal date, with the amendment being implemented within 90 days of approval.

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In accordance with 10 CFR 50.91, NMC is providing a copy of this letter and enclosures to each facility's designated State Official.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on February 16, 2006.

Edward J. Weinkam
Director, Nuclear Licensing and Regulatory Services
Nuclear Management Company, LLC

Enclosures (13)

cc: Administrator, Region III, USNRC
Project Manager, Palisades Nuclear Plant, Point Beach Nuclear Plant, and Prairie Island Nuclear Generating Plant, USNRC
Senior Resident Inspector, Palisades Nuclear Plant, Point Beach Nuclear Plant, and Prairie Island Nuclear Generating Plant, USNRC
State Official, Lou Brandon – Chief – NFU/HWRS/WHMD, Ms. Ave M. Bie – Public Service Commission of WI, Minnesota Department of Commerce

ENCLOSURE 1

Description and Assessment

1.0 INTRODUCTION

The proposed license amendment revises the requirements in Technical Specifications (TS) related to steam generator tube integrity. The changes are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. The availability of this technical specification improvement was announced in the Federal Register (FR) on May 6, 2005 as part of the consolidated line item improvement process (CLIIP).

2.0 DESCRIPTION OF PROPOSED AMENDMENT

Consistent with the NRC-approved Revision 4 of TSTF-449, the proposed TS changes include (Each facility's unique TS Section identification is provided in Table 1 below and exceptions, if any, are provided in enclosure 2):

- Revised TS definition of LEAKAGE
- Revised TS, "RCS [Reactor Coolant System] Operational Leakage"
- New TS, "Steam Generator (SG) Tube Integrity"
- Revised TS, "Steam Generator (SG) Program"
- Revised TS, "Steam Generator Tube Inspection Report"

Proposed revisions to the TS Bases are also included in this application. As noted in Enclosure 2 for each facility, the TSTF-449, Revision 4 approved Bases have been modified to incorporate plant specific analyses and TS requirements. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4 is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126) the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

5.0 TECHNICAL ANALYSIS

The Nuclear Management Company, LLC (NMC) has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR 10298) as part of the CLIIP Notice for Comment. This included the NRC staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. NMC has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to each of the facilities identified in this license amendment request and justify this amendment for the incorporation of the changes to each facility's TS. Clarifications for each facility are identified in Enclosure 2 for the TS and Bases which incorporate plant specific analyses and TS requirements.

6.0 REGULATORY ANALYSIS

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

6.1 Verification and Commitments

The information in Enclosure 3 is provided to support the NRC staff's review of this amendment application.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

NMC has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. NMC has concluded that the proposed determination presented in the notice is applicable to each of the facilities identified in this license amendment request and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

8.0 ENVIRONMENTAL EVALUATION

NMC has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. NMC has concluded that the staff's findings presented in that evaluation are applicable to each of the facilities identified in this license amendment request and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. NMC is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4 (except as noted in Sections 2 and 5), or the NRC staff's model SE published on March 2, 2005 (70 FR 10298). However, unique characteristics of each facility's TS and Bases in relationship to TSTF-449 are identified in Enclosure 2. The differences between each facility's proposed TS and TSTF-449 do not affect the no significant hazards consideration determination and environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP.

10.0 REFERENCES

Federal Register Notices:

Notice for Comment published on March 2, 2005 (70 CFR 10298)
Notice of Availability published on May 6, 2005 (70 FR 24126)

Table 1

Facility Unique TS Section

TSTF-449 TS Section Description	Palisades Nuclear Plant	Point Beach Nuclear Plant Units 1 and 2	Prairie Island Nuclear Generating Plant Units 1 and 2
Definition of LEAKAGE	1.1	1.1	1.1
RCS [Reactor Coolant System] Operational Leakage[1]	3.4.13	3.4.13	3.4.14
Steam Generator (SG) Tube Integrity	3.4.17	3.4.17	3.4.19
Steam Generator (SG) Program	5.5.8	5.5.8	5.5.8
Steam Generator Tube Inspection Report	5.6.8	5.6.8	5.6.7

1 PCS [Primary Coolant System] Operational Leakage in Palisades Nuclear Plant Technical Specifications

ENCLOSURE 2

The following Plant Specific Clarifications Of TSTF-449 With Respect To Each Facility's Technical Specifications and Bases are contained within Enclosure 2:

Enclosure 2A – Palisades Nuclear Plant

Enclosure 2B – Point Beach Nuclear Plant Units 1 and 2

Enclosure 2C – Prairie Island Nuclear Generating Plant Units 1 and 2

ENCLOSURE 2A

Plant Specific Clarifications Of TSTF-449 With Respect To Each Facility's Technical Specifications and Bases

Palisades Nuclear Plant (PNP)

PNP TS/Bases	ISTS[1]	Location	Description of TS/Bases	Basis
3.4.13	3.4.13	LCO[2] statement	No changes proposed to remove 1 gpm primary to secondary LEAKAGE	PNP TS is currently consistent with TS as described in TSTF-449 and no change is required
B 3.4.13	B 3.4.13	LCO discussion	No changes proposed to remove 1 gpm primary to secondary LEAKAGE	PNP TS do not currently include 1 gpm and no change is required

1. NUREG-1432, Standard Technical Specifications, Combustion Engineering Plants
2. Limiting Condition for Operation

ENCLOSURE 2B

Plant Specific Clarifications Of TSTF-449 With Respect To Each Facility's Technical Specifications and Bases

Point Beach Nuclear Plant (PBNP)

PBNP TS/Bases	ISTS[1]	Location	Description of TS/Bases	Basis
3.4.13	3.4.13	LCO[2] statement	No changes proposed to remove 1 gpm primary to secondary LEAKAGE	PBNP TS is currently consistent with TS as described in TSTF-449 and no change is required
5.5.8	5.5.9	SG Program	Included two SG tube inspection paragraphs in 5.5.8.d.2	Unit 2 SG tubes are different materials than Unit 1 SG, thus different inspection requirements are proposed for each unit
B 3.4.13	B 3.4.13	ASA[3] discussion	Discusses accident analyses based on primary to secondary leakage per SG	Plant specific analyses are based on per SG limit
B 3.4.13	B 3.4.13	LCO discussion	No changes proposed to remove 1 gpm primary to secondary LEAKAGE	PBNP TS do not currently include 1 gpm and no change is required
B 3.4.13	B 3.4.13	ASA and LCO discussion	Discusses accident analyses based on primary to secondary leakage per SG	Plant specific analyses are based on per SG limit

1. NUREG-1431, Standard Technical Specifications, Westinghouse Plants
2. Limiting Condition for Operation
3. Applicable Safety Analyses

ENCLOSURE 2C

Plant Specific Clarifications Of TSTF-449 With Respect To Each Facility's Technical Specifications and Bases

Prairie Island Nuclear Generating Plant (PINGP)

PINGP TS/Bases	ISTS[1]	Location	Description of TS/Bases	Basis
3.4.14	3.4.13	LCO[2] statement	No changes proposed to remove 1 gallons per minute primary to secondary LEAKAGE, add 150 gallons per day	PINGP TS is currently consistent with TS as described in TSTF-449 and no change is required
3.4.14	3.4.13	Conditions A and B	PINGP made changes similar to TSTF-449 in 3.4.14 Conditions C and D	Unique PINGP TS requirements
5.6.7	5.6.9	Paragraph b	Included PINGP specific report requirements for implementation of voltage-based repair criteria to tube support plate intersections	Current TS requirements
B 3.4.5	B 3.4.5	LCO discussion	No change	TSTF change not applicable due to unique PINGP TS requirements
B 3.4.6	B 3.4.6	LCO discussion	No change	TSTF change not applicable due to unique PINGP TS requirements
B 3.4.7	B 3.4.7	LCO discussion	No change	TSTF change not applicable due to unique PINGP TS requirements
B 3.4.14	B 3.4.13	ASA[3] discussion	Discusses PINGP SGTR[4] and SLB[5] accident analyses	Plant specific information

PINGP TS/Bases	ISTS[1]	Location	Description of TS/Bases	Basis
B 3.4.14	B 3.4.13	LCO discussion	No changes proposed to remove 1 gpm primary to secondary LEAKAGE	PINGP TS do not currently include 1 gpm and no change is required
B 3.4.14	B 3.4.13	Conditions A and B discussion	PINGP made changes similar to TSTF-449 in 3.4.14 Conditions C and D	Unique PINGP TS requirements

1. NUREG-1431, Standard Technical Specifications, Westinghouse Plants
2. Limiting Condition for Operation
3. Applicable Safety Analyses
4. Steam Generator Tube Rupture
5. Steam Line Break

ENCLOSURE 3

The following Unit Specific Steam Generator Information is contained within Enclosure 3:

Enclosure 3A – Palisades Nuclear Plant

Enclosure 3B – Point Beach Nuclear Plant Units 1 and 2

Enclosure 3C – Prairie Island Nuclear Generating Plant Units 1 and 2

ENCLOSURE 3A

Unit Specific Steam Generator Information

Palisades Nuclear Plant

Required Steam Generator (SG) Information	**Palisades Nuclear Plant**	
Steam Generator (SG) Model(s):	Combustion Engineering CE 2530	
Effective Full Power Years (EFPY) of service for currently installed SGs	11.5 (Through cycle 18)	
Tubing Material (e.g., 600M, 600TT, 660TT)	600 Mill Annealed	
Number of tubes per SG	8219	
Number and percentage of tubes plugged in each SG	SG A	SG B
	380	363
	4.62 %	4.42 %
Number of tubes repaired in each SG	SG A	SG B
	0	0
Degradation mechanism(s) identified	ODSCC top of tubesheet, eggcrates, dents/dings PWSCC tubesheet, eggcrates Wear vertical straps, diagonal bars and eggcrates Wear from loose parts	

Required Steam Generator (SG) Information	**Palisades Nuclear Plant**
Current primary -to-secondary leakage limits: per SG; Total; Leakage is evaluated at what temperature condition?	0.3 gallons per minute per SG, 0.3 gallons per minute total; leakage evaluated at Primary Coolant System (PCS) normal operating temperatures
Approved Alternate Tube Repair Criteria (ARC): (Provide for each) Approved by [amendment number dated ___]; Applicability (e.g., degradation mechanism, location); any special limits on allowable accident leakage; any exceptions or clarifications to the structural performance criteria that apply to the ARC	None
Approved SG Tube Repair Methods (Provide for each): Approved by [amendment number dated ___]; Applicability limits, if any; Sleeve repair criteria (e.g., 40% of the initial sleevewall thickness)	None
Performance criteria for accident leakage (Primary to secondary leak rate values assumed in licensing basis accident analysis, including assumed temperature conditions)	0.3 gallons per minute per at PCS normal operating temperatures

ENCLOSURE 3B

Unit Specific Steam Generator Information

Point Beach Nuclear Plant Units 1 and 2 (PBNP)

Required Steam Generator (SG) Information	**PBNP Unit 1**		**PBNP Unit 2**	
Steam Generator (SG) Model(s):	Westinghouse Series 44F		Westinghouse Series D47F	
Effective Full Power Years (EFPY) of service for currently installed SGs	17.7 at U1R29 (Replaced 10/1983)		6.4 at U2R27 (Replaced 10/1996)	
Tubing Material (e.g., 600M, 600TT, 660TT)	600 Thermally Treated		690 Thermally Treated	
Number of tubes per SG	3214		3499	
Number and percentage of tubes plugged in each SG	A SG	B SG	A SG	B SG
	4	6	0	4
	0.1%	0.2%	0%	0.1%
Number of tubes repaired in each SG	A SG	B SG	A SG	B SG
	0	0	0	0
Degradation mechanism(s) identified	None except minor anti-vibration bar and cold leg support wear		None	

Required Steam Generator (SG) Information	**PBNP Unit 1**	**PBNP Unit 2**
Current primary -to-secondary leakage limits: per SG; Total; Leakage is evaluated at what temperature condition?	500 gallons per day per SG; 1000 gallons per day total; leakage is evaluated at Reactor Coolant System (RCS) operating temperature (Tave)	500 gallons per day per SG; 1000 gallons per day total; leakage is evaluated at RCS operating temperature (Tave)
Approved Alternate Tube Repair Criteria (ARC): (Provide for each) Approved by [amendment number dated ___]; Applicability (e.g., degradation mechanism, location); any special limits on allowable accident leakage; any exceptions or clarifications to the structural performance criteria that apply to the ARC	None	None
Approved SG Tube Repair Methods (Provide for each): Approved by [amendment number dated ___]; Applicability limits, if any; Sleeve repair criteria (e.g., 40% of the initial sleevewall thickness)	None	None
Performance criteria for accident leakage (Primary to secondary leak rate values assumed in licensing basis accident analysis, including assumed temperature conditions)	0.35 gallons per minute per SG at RCS operating temperature (Tave)	0.35 gallons per minute per SG at RCS operating temperature (Tave)

ENCLOSURE 3C

Unit Specific Steam Generator Information

Prairie Island Nuclear Generating Plant Units 1 and 2 (PINGP)

Required Steam Generator (SG) Information	**PINGP Unit 1**		**PINGP Unit 2**	
Steam Generator (SG) Model(s):	Framatome ANP Model 56/19		Westinghouse Model 51	
Effective Full Power Years (EFPY) of service for currently installed SGs	1 (Replaced 11/2004)		26.1 (through Cycle 22)	
Tubing Material (e.g., 600M, 600TT, 660TT)	690 Thermally Treated		600 Mill Annealed	
Number of tubes per SG	4868		3388	
Number and percentage of tubes plugged in each SG	11 SG	12 SG	21 SG	22 SG
	0	0	242	258
	0 %	0 %	7.14%	7.62%
Number of tubes repaired in each SG	11 SG	12 SG	21 SG	22 SG
	0	0	1274	774

Required Steam Generator (SG) Information	**PINGP Unit 1**	**PINGP Unit 2**
Degradation mechanism(s) identified	None	Primary water stress corrosion cracking, secondary side intergranular and stress corrosion cracking and wear due to loose parts, cold leg thinning at tube support plates (TSP), wear at antivibration bars.
Current primary -to-secondary leakage limits: per SG; Total; Leakage is evaluated at what temperature condition?	150 gallons per day per SG; 300 gallons per day total; leakage evaluated at room temperature	150 gallons per day per SG; 300 gallons per day total; leakage evaluated at room temperature
Approved Alternate Tube Repair Criteria (ARC): (Provide for each) Approved by [amendment number dated ___]; Applicability (e.g., degradation mechanism, location); any special limits on allowable accident leakage; any exceptions or clarifications to the structural performance criteria that apply to the ARC	None are applicable to the Replacement Steam Generators. The existing Prairie Island Alternate Tube Repair Criteria apply to only Westinghouse Model 51 Steam Generators (Unit 2 steam generators)	1. F* Steam Generator Tube Repair Criteria: License Amendment (LA) – 118/111 dated May 15, 1995; Applicable to all degradation mechanisms below the F* hard roll; Due to tubesheet flexure assumptions in WCAP-14225, the uppermost location height of the top of the F* hard roll distance is the middle of the tubesheet. The middle of the tubesheet is 10.72 inches above the tube end. Acceptable distance (not including eddy current measurement uncertainty) is 1.07 inches; Site specific leakages are assigned to each F* tube and included in the total main steam line break (MSLB) leakage for all degradation mechanisms for the operational assessment.; No special limits on allowable accident leakage and no clarification to the structural performance criteria.

Required Steam Generator (SG) Information	PINGP Unit 1	PINGP Unit 2
		2. Voltage Based, LA – 133/125 dated November 18, 1997; applies to degradation due to predominantly axially oriented outside diameter stress corrosion cracking confined within the tube to tube support plate locations; Indication specific leakages are assigned per Generic Letter 95-05 and Nuclear Energy Institute follow-on guidance for each indication and included in the total MSLB leakage for all degradation mechanisms for the operational assessment.; special limit on allowable primary to secondary MSLB accident leakage of 1.42 gallons per minute (at 578 ºF); no clarification to the structural performance criteria. 3. EF* SG alternate repair criteria, LA - 137/128 dated August 13, 1998 and LA – 149/140; Due to tubesheet flexure assumptions in WCAP-14225, the uppermost location height of the top of the EF* hard roll distance is 2 inches from the top of the tubesheet. The top of the tubesheet is 21.44 inches above the tube end. Acceptable distance (not including eddy current measurement uncertainty) is 1.67 inches above the tube end; Site specific leakages are assigned to each EF* tube and included in the total MSLB leakage for all degradation mechanisms for the operational assessment.; No

Required Steam Generator (SG) Information	PINGP Unit 1	PINGP Unit 2
		special limits on allowable accident leakage and no clarification to the structural performance criteria.
Approved SG Tube Repair Methods (Provide for each): Approved by [amendment number dated ___]; Applicability limits, if any; Sleeve repair criteria (e.g., 40% of the initial sleevewall thickness)	None	1. a. Tube sleeving; LA – 76/69 dated October 11, 1985 (superceded by LA 132/124); Tubesheet Sleeves, 50%. b. Welded sleeving improvements; LA – 132/124 dated November 4, 1997; Tubesheet and TSP locations, Sleeve repair criteria, 31%. c. Incorporate Combustion Engineering Topical Report CEN 629-P, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves," Revision 3 Repair criteria, LA – 144/135 dated April 15, 1999; Applicable to Sleeve Joints, 25%. 2. Additional Roll Expansion (F* reroll): LA-118/111 dated May 15, 1995; incorporate Westinghouse report WCAP-14225, "F* and L* Plugging Criteria for Tubes with Degradation in the Tubesheet Roll Expansion Region of the Prairie Island Units 1 and 2 Steam Generators", the basis document for rerolling is Combustion Engineering CEN-620-P; Applicable only below midplane of the tubesheet. Reroll must satisfy F* criteria.

Required Steam Generator (SG) Information	**PINGP Unit 1**	**PINGP Unit 2**
		3. Additional Roll Expansion (EF* reroll): LA-137/128 dated August 13, 1998 and LA – 149/140; incorporate Westinghouse report WCAP-14255, Revision 2, "F* and Elevated F* Tube Plugging Criteria for Tubes with Degradation in the Tubesheet Region of the Prairie Island Units 1 and 2 Steam Generators", the basis document for rerolling is Combustion Engineering CEN-620-P; Applicable anywhere below 2 inches from the top of the tubesheet which allows use of the EF* criteria.
Performance criteria for accident leakage (Primary to secondary leak rate values assumed in licensing basis accident analysis, including assumed temperature conditions)	1.0 gallon per minute at 70 °F	1.0 gallon per minute at 70 °F

ENCLOSURE 4

The following Proposed Technical Specification and Bases Pages (markup) are contained within Enclosure 4:

Enclosure 4A – Palisades Nuclear Plant

Enclosure 4B – Point Beach Nuclear Plant Units 1 and 2

Enclosure 4C – Prairie Island Nuclear Generating Plant Units 1 and 2

ENCLOSURE 4A

Proposed Technical Specification and Bases Pages (markup)

Palisades Nuclear Plant

Technical Specification Pages

1.1-4	5.0-14
3.4.13-1	5.0-15
3.4.13-2	5.0-16
3.4.17-1	5.0-17
3.4.17-2	5.0-18
5.0-11	5.0-30
5.0-12	5.0-31
5.0-13	

Bases pages

B 3.4.4-2	B 3.4.13-7
B 3.4.5-3	B 3.4.17-1
B 3.4.6-3	B 3.4.17-2
B 3.4.7-4	B 3.4.17-3
B 3.4.13-2	B 3.4.17-4
B 3.4.13-3	B 3.4.17-5
B 3.4.13-4	B 3.4.17-6
B 3.4.13-5	B 3.4.17-7
B 3.4.13-6	

32 pages follow

1.1 Definitions

LEAKAGE

a. Identified LEAKAGE (continued)

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

3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator ~~(SG)~~ to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE (except Primary Coolant Pump seal leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary ~~SG~~ LEAKAGE) through a nonisolable fault in an PCS component body, pipe wall, or vessel wall.

MODE

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

OPERABLE - OPERABILITY

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

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.13 PCS Operational LEAKAGE

LCO 3.4.13 PCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150~~432~~ gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary leakage.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	------------------------------NOTES------------------------ 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. -- Verify PCS operational LEAKAGE is within limits by performance of PCS water inventory balance.	----------NOTE-------- Only required to be performed during steady state operation -------------------------- 72 hours
SR 3.4.13.2	----------------------------- NOTE ------------------------- Not required to be performed until 12 hours after establishment of steady state operation. -- Verify ~~SG tube integrity is in accordance with the Steam Generator Tube Surveillance Program~~primary to secondary LEAKAGE is < 150 gallons per day through any one SG.	72 hours ~~In accordance with the Steam Generator Tube Surveillance Program~~

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.7 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 (B&PV Code) as follows:

B&PV Code terminology for inservice testing activities	Required interval for performing inservice testing activities
Weekly	≤ 7 days
Monthly	≤ 31 days
Quarterly or every 3 months	≤ 92 days
Semiannually or every 6 months	≤ 184 days
Every 9 months	≤ 276 days
Yearly or annually	≤ 366 days
Biennially or every 2 years	≤ 731 days

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

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

d. Nothing in the B&PV Code shall be construed to supersede the requirements of any Technical Specification.

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.3 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "PCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

5.5 Programs and Manuals

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

~~This program provides controls for surveillance testing of the Steam Generator (SG) tubes to ensure that the structural integrity of this portion of the Primary Coolant System (PCS) is maintained. The program shall contain controls to ensure:~~

~~a. Steam Generator Tube Sample Selection and Inspection~~

~~The inservice inspection may be limited to one SG on a rotating schedule encompassing 6% of the tubes if the results of previous inspections indicate that both SGs are performing in a like manner. If the operating conditions in one SG are found to be more severe than those in the other SG, the sample sequence shall be modified to inspect the most severe conditions.~~

5.5 Programs and Manuals

~~5.5.8 Steam Generator Tube Surveillance Program~~

~~a. Steam Generator Tube Sample Selection and Inspection (continued)~~

~~The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.8-1. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:~~

~~1. 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.~~

~~2. The first sample of tubes selected for each inservice inspection of each SG shall include:~~

~~a) All nonplugged tubes that previously had detectable wall penetrations greater than 20%.~~

~~b) Tubes in those areas where experience has indicated potential problems.~~

~~c) A tube inspection shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~

~~3. The tubes selected as the second and third samples (if required by Table 5.5.8-1) during each inservice inspection may be subjected to a partial tube inspection provided:~~

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

~~b) The inspections include those portions of the tubes where imperfections were previously found.~~

5.5 Programs and Manuals

~~5.5.8 Steam Generator Tube Surveillance Program~~

~~a. Steam Generator Tube Sample Selection and Inspection (continued)~~

~~4. 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 must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.~~

~~b. Inspection Frequencies~~

~~The above required inservice inspection of SG tubes shall be performed at the following frequencies:~~

~~1. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspections 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.~~

5.5 Programs and Manuals

~~5.5.8 Steam Generator Tube Surveillance Program~~

~~b. Inspection Frequencies (continued)~~

~~2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.5.8-1 at 40 month intervals fall into 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 5.5.8.b.1; the interval may then be extended to a maximum of once per 40 months.~~

~~3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.8-1 during the shutdown subsequent to any of the following conditions:~~

~~a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of LCO 3.4.13.~~

~~b) A seismic occurrence greater than the Operating Basis Earthquake.~~

~~c) A loss-of-coolant accident resulting in initiation of flow of the engineered safeguards.~~

~~d) A main steam line or main feedwater line break.~~

~~c. Acceptance Criteria~~

~~1. As used in this Specification:~~

~~a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.~~

~~b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.~~

~~c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.~~

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program

~~b. Inspection Frequencies (continued)~~

~~d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.~~

~~e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.~~

~~f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.~~

~~g) Unserviceable described the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.8.b.3, above.~~

~~h) Tube Inspection means an inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.~~

~~i) Preservice Inspection means an inspection of the full length of each tube in SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the shop hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.~~

~~2. The SG shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.8-1.~~

~~The provisions of SR 3.0.2 are applicable to the Steam Generator Tube Surveillance Program.~~

~~TABLE 5.5.8-1~~
~~STEAM GENERATOR TUBE INSPECTION~~

~~1ST SAMPLE INSPECTION~~[1]		~~2ND SAMPLE INSPECTION~~		~~3RD SAMPLE INSPECTION~~	
~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~C-1~~	~~None~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
~~C-2~~	~~Plug defective tubes and inspect additional 2S tubes in this SG.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
~~C-3~~	~~Inspect all tubes in this SG, plug defective tubes and inspect 2S tubes in each other SG.~~	~~C-2~~	~~Plug defective tubes and inspect additional 4S tubes in this SG.~~	~~C-1~~	~~None~~
				~~C-2~~	~~Plug defective tubes~~
				~~C-3~~	~~Perform action for C-3 result of first Sample~~
		~~C-3~~	~~Perform action for C-3 result of first Sample~~	~~N/A~~	~~N/A~~
		~~All other SGs are C-1~~	~~None~~	~~N/A~~	~~N/A~~
		~~Some SGs C-2 but no other SG is C-3~~	~~Perform action for C-2 result of second sample~~	~~N/A~~	~~N/A~~
		~~Other SG is C-3~~	~~Inspect all tubes each SG and plug defective tubes~~	~~N/A~~	~~N/A~~

~~NOTES: 1 The minimum sample size for the first sample inspection is S tubes per SG where S=(6/n)%, where n is the number of steam generators inspected during an inspection.~~

5.6 Reporting Requirements

5.6.6 Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring 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 to OPERABLE status.

5.6.7 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Dome Delamination tests within 90 days after completion of the tests.

5.6.8 Steam Generator Tube Inspection~~Surveillance~~ Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

~~The following reports shall be submitted to the Commission following each inservice inspection of steam generator tubes:~~

~~a. The number of tubes plugged in each steam generator shall be reported to the Commission within 15 days following the completion of each inspection, and~~

5.6 Reporting Requirements

~~b. The complete results of the steam generator tube inservice inspection shall be reported to the Commission within 12 months following completion of the inspection. This report shall include:~~

~~1. Number and extent of tubes inspected.~~

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

~~3. Identification of tubes plugged.~~

~~c. Results of steam generator tube inspections that fall into Category C-3 shall require 24 hour verbal notification to the NRC prior to resumption of plant operation. A written followup within the next 30 days shall provide a description of investigations and corrective measures taken to prevent recurrence.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four PCPs are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to PCP operation are the Loss of Forced Primary Coolant Flow, Primary Coolant Pump Rotor Seizure and Uncontrolled Control Rod Withdrawal events (Ref. 1).

Steady state DNB analysis had been performed for the four pump combination. The steady state DNB analysis, which generates the pressure and temperature and Safety Limit (i.e., the Departure from Nucleate Boiling Ratio (DNBR) limit), assumes a maximum power level of 110.4% RTP. This is the design overpower condition for four pump operation. The 110.4% value is the accident analysis setpoint of the trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

PCS Loops - MODES 1 and 2 satisfy Criteria 2 and 3 of 10 CFR 50.36(c)(2).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both PCS loops with both PCPs in each loop in operation for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

Each OPERABLE loop consists of two PCPs providing forced flow for heat transport to an SG that is OPERABLE~~ in accordance with the Steam Generator Tube Surveillance Program~~. SG, and hence PCS loop OPERABILITY with regards to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in MODE 3 if any SG water level is ≤ 25.9% (narrow range) as sensed by the RPS. The minimum level to declare the SG OPERABLE is 25.9% (narrow range).

In MODES 1 and 2, the reactor can be critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all PCS loops are required to be in operation in these MODES to prevent DNB and core damage.

BASES

LCO (continued)

d. SG secondary temperature is < 100 °F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

Satisfying any of the above conditions will preclude a large pressure surge in the PCS when the PCP is started. Energy additions from the steam generators could occur if a PCP was started when the steam generator secondary temperature is significantly above the PCS temperature. The maximum pressurizer level at which credit is taken for having a bubble (57%, which provides about 700 cubic feet of steam space) is based on engineering judgement and verified by LTOP analysis. This level provides the same steam volume to dampen pressure transients as would be available at full power.

An OPERABLE PCS loop consists of any one (of the four) OPERABLE PCP and an SG that is OPERABLE~~ in accordance with the Steam Generator Tube Surveillance Program~~ and has the minimum water level specified in SR 3.4.5.2. A PCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one PCS loop in operation is adequate for transport and heat removal. A second PCS loop is required to be OPERABLE but is not required to be in operation for redundant heat removal capability.

Operation in other MODES is covered by:

LCO 3.4.4, "PCS Loops-MODES 1 and 2";

LCO 3.4.6, "PCS Loops-MODE 4";

LCO 3.4.7, "PCS Loops-MODE 5, Loops Filled";

LCO 3.4.8, "PCS Loops-MODE 5, Loops Not Filled";

LCO 3.9.4, "Shutdown Cooling (SDC) and Coolant Circulation-High Water Level" (MODE 6); and

LCO 3.9.5, "Shutdown Cooling (SDC) and Coolant Circulation-Low Water Level" (MODE 6)

BASES

LCO (continued)

Note 2 requires that one of the following conditions be satisfied before forced circulation (starting the first PCP) may be started:

a. SG secondary temperature is $\leq T_c$;

b. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and PCS heatup/cooldown rate is ≤ 10°F/hour; or

c. SG secondary temperature is < 100°F above T_c, and shutdown cooling is isolated from the PCS, and pressurizer level is ≤ 57%.

Satisfying any of the above conditions will preclude a large pressure surge in the PCS when the PCP is started. Energy additions from the steam generators could occur if a PCP was started when the steam generator secondary temperature is significantly above the PCS temperature. The maximum pressurizer level at which credit is taken for having a bubble (57%, which provides about 700 cubic feet of steam space) is based on engineering judgement and verified by LTOP analysis. This level provides the same steam volume to dampen pressure transients as would be available at full power.

Note 3 specifies a limitation on the simultaneous operation of primary coolant pumps P-50A and P-50B which allows the pressure limits in LCO 3.4.3, "PCS Pressure and Temperature Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection System," to be higher than they would be without this limit. This is because the pressure in the reactor vessel downcomer region when primary coolant pumps P-50A and P-50B are operated simultaneously is higher than the pressure for other two primary coolant pump combinations.

An OPERABLE PCS loop consists of any one (of the four) OPERABLE PCP and an SG that has the minimum water level specified in SR 3.4.6.2 and is OPERABLE~~ in accordance with the Steam Generator Tube Surveillance Program~~. PCPs are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

BASES

LCO (continued)

Satisfying any of the above conditions will preclude a large pressure surge in the PCS when the PCP is started. Energy additions from the steam generators could occur if a PCP was started when the steam generator secondary temperature is significantly above the PCS temperature. The maximum pressurizer level at which credit is taken for having a bubble (57%, which provides about 700 cubic feet of steam space) is based on engineering judgement and verified by LTOP analysis. This level provides the same steam volume to dampen pressure transients as would be available at full power.

Note 4 specifies a limitation on the simultaneous operation of primary coolant pumps P-50A and P-50B which allows the pressure limits in LCO 3.4.3, "PCS Pressure and Temperature Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection System," to be higher than they would be without this limit.

Note 5 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting SDC trains to not be in operation when at least one PCP is in operation. This Note provides for the transition to MODE 4 where a PCP is permitted to be in operation and replaces the PCS circulation function provided by the SDC trains.

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger. SDC pumps are OPERABLE if they are capable of being powered and are able to provide forced flow through the reactor core.

An SG can perform as a heat sink via natural circulation when:

a. SG has the minimum water level specified in SR 3.4.7.2.

b. SG is OPERABLE ~~in accordance with the SG Tube Surveillance Program.~~

c. SG has available method of feedwater addition and a controllable path for steam release.

d. Ability to pressurize and control pressure in the PCS.

If both SGs do not meet the above provisions, then LCO 3.4.7 item b (i.e. the secondary side water level of each SG shall be $\geq$ -84%) is not met.

BACKGROUND (continued)

As defined in 10 CFR 50.2, the PCPB includes all those pressure-containing components, such as the reactor pressure vessel, piping, pumps, and valves, which are:

(1) Part of the primary coolant system, or

(2) Connected to the primary coolant system, up to and including any and all of the following:

  (i) The outermost containment isolation valve in system piping which penetrates the containment,

  (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate the containment,

  (iii) The pressurizer safety valves and PORVs.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for all events resulting in a discharge of steam from the steam generators to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 0.3 gpm or increases to 0.3 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis~~0. 3 gpm primary to secondary LEAKAGE as the initial condition~~.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR) and the Control Rod Ejection (CRE) accident analyses. The leakage contaminates the secondary fluid.

The FSAR (Ref. 2 and 5) analysis for SGTR assumes the contaminated secondary fluid is released via the Main Steam Safety Valves and Atmospheric Dump Valves. The 0.3 gpm primary to secondary LEAKAGE safety analysis assumption is inconsequential, relative to the dose contribution from the affected SG.

The MSLB (Ref 3 and 5) is more limiting than SGTR for site radiation releases. The safety analysis for the MSLB accident assumes the entire 0.3 gpm primary to secondary LEAKAGE is through the affected~~in one~~ steam generator as an initial condition.

The CRE (Ref 4 and 5) accident with primary fluid release through the Atmospheric Dump Valves is the most limiting event for site radiation releases. The safety analysis for the CRE accident assumes 0.3 gpm primary to secondary LEAKAGE in one steam generator as an initial condition.

The dose consequences resulting from the SGTR, MSLB and CRE accidents are well within the guidelines defined in 10 CFR 100 and meets the requirements of Appendix A of 10 CFR 50 (GDC 19).

PCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

PCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

As defined in Section 1.0, pressure boundary LEAKAGE is "LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an PCS component body, pipe wall, or vessel wall."

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE from within the PCPB 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 PCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE from within the PCPB is allowed because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the PCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically located sources which is known not to adversely affect the OPERABILITY of required leakage detection systems, but does not include pressure boundary LEAKAGE or controlled Primary Coolant Pump (PCP) seal leakoff to the Volume Control Tank (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

LCO 3.4.14, "PCS 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 PCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the PCS, the loss must be included in the allowable identified LEAKAGE.

LCO (continued)

d. Primary to Secondary LEAKAGE ~~T~~through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 6). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures. ~~The 432 gallons per day limit on primary to secondary LEAKAGE through any one SG ensures the total primary to secondary LEAKAGE through both SGs produces acceptable offsite doses in the MSLB accident analysis. In addition, the LEAKAGE limit also ensures that SG integrity is maintained in the event of a CRE, MSLB or under LOCA conditions. Violation of this LCO could exceed the offsite dose limits for this accident analysis. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

APPLICABILITY

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

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

ACTIONS

A.1

Unidentified LEAKAGE or~~,~~ 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 PCPB.

B.1 and B.2

If any pressure boundary LEAKAGE from within the PCPB exists or primary to secondary LEAKAGE is not within limit, or if unidentified~~,~~ or identified~~, 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. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

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

The PCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. The Surveillance is modified by two Notes. ~~Therefore, this SR is modified by a~~ Note 1 ~~which~~ states that the SR is not required to be performed in MODES 3 and 4, until 12 hours of steady state operation have elapsed.

Steady state operation is required to perform a proper water inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met only when steady state is established. For PCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable PCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and PCP seal leakoff.

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

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

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

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 7. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 7).
~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.~~

REFERENCES

1. FSAR, Section 5.1.5

2. FSAR, Section 14.15

3. FSAR, Section 14.14

4. FSAR, Section 14.16

5. FSAR, Section 14.24

6. NEI 97-06, "Steam Generator Program Guidelines."

7. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 PRIMARY COOLANT SYSTEM (PCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the primary coolant pressure boundary (PCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the PCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the PCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "PCS Loops - MODES 1 and 2," LCO 3.4.5, "PCS Loops - MODE 3," LCO 3.4.6, "PCS Loops - MODE 4," and LCO 3.4.7, "PCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended PCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.8. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES *(continued)*

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "PCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via the Main Steam Safety Valves and Atmospheric Dump Valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.3 gpm or is assumed to increase to 0.3 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "PCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall , between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 0.3 gpm per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "PCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit 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 break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

PCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

BASES

ACTIONS

A.1 and A.2 (continued)

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1 (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

[The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES (continued)

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

ENCLOSURE 4B

Proposed Technical Specification and Bases Pages (markup)

Point Beach Nuclear Plant Units 1 and 2

Technical Specification Pages

1.1-3	5.5-10
3.4.13-1	5.5-11
3.4.13-2	5.5-12
3.4.17-1	5.5-13
3.4.17-2	5.5-14
5.5-7	5.6-6
5.5-8	5.6-7
5.5-9	

Bases pages

B 3.4.4-2	B 3.4.13-7
B 3.4.5-3	B 3.4.17-1
B 3.4.6-2	B 3.4.17-2
B 3.4.7-3	B 3.4.17-3
B 3.4.13-2	B 3.4.17-4
B 3.4.13-3	B 3.4.17-5
B 3.4.13-4	B 3.4.17-6
B 3.4.13-5	B 3.4.17-7
B 3.4.13-6	

32 pages follow

1.1 Definitions

L_a

The maximum allowable primary containment leakage rate, L_a, shall be 0.4% of primary containment air weight per day at the peak design containment pressure (P_a).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

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

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

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary ~~SG~~ LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

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; and

d. 150~~500~~ gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	---------------------------NOTES------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS Operational LEAKAGE~~eakage~~ is within limits by performance of RCS water inventory balance.	 72 hours
SR 3.4.13.2	---------------------------NOTE--------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify ~~steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program~~ primary to secondary LEAKAGE is < 150 gallons per day through any one SG.	~~In accordance with the Steam Generator Tube Surveillance Program~~ 72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of the leakage rate for

5.5 Programs and Manuals

an individual SG. Leakage is not to exceed 500 gallons per day per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. i. Unit 1 (alloy 600 Thermally Treated tubes): Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

   ii. Unit 2 (alloy 690 Thermally Treated tubes): Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling

5.5 Programs and Manuals

outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

~~This program provides controls for inservice inspection and testing of steam generator tubing.~~

~~a. Definitions.~~

~~1. Tube Inspection – Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered a tube inspection.~~

~~2. Imperfection – An exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.~~

~~3. Degradation – A service induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.~~

~~4. Degraded Tube – A tube that contains imperfections caused by degradation greater than 20% of the nominal tube wall thickness.~~

~~5. Defect – An imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.~~

~~6. Plugging Limit – The imperfection depth beyond which the tube must be removed from service or repaired, because the tube~~

5.5 Programs and Manuals

~~may become defective prior to the next scheduled inspection.~~
~~The plugging limit is 40% of the nominal wall thickness.~~

5.5 Programs and Manuals

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~b. Sample Selection and Testing~~

~~Selection and testing of steam generator tubes shall be made on the following basis:~~

~~1. One steam generator of each unit may be selected for inspection during inservice inspection in accordance with the following requirements:~~

~~i. The inservice inspection may be limited to one steam generator on an alternating sequence basis. This examination shall include at least 6% of the tubes if the results of the first or a prior inspection indicate that both generators are performing in a comparable manner.~~

~~ii. When both steam generators are required to be examined by Table 5.5.8-1 and if the condition of tubes in one steam generator is found to be more severe than in the other steam generator of a unit, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator with the more severe condition.~~

~~2. The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements of Table 5.5.8-1. The results of each sampling examination of a steam generator shall be classified in the following three categories:~~

~~i. Category C-1: Less than 5% of the total tubes examined are degraded but none are defective.~~

~~ii. Category C-2: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective or one tube to not more than 1% of the sample is defective.~~

~~iii. Category C-3: More than 10% of the total number of tubes examined are degraded, but none are defective or more than 1% of the sample is defective.~~

5.5 Programs and Manuals

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~If the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the tube nominal wall thickness.~~

~~3.Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.~~

~~4.In addition to the sample size specified in Table 5.5.8-1, the tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded.~~

~~5.During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where imperfections were detected during the prior examination.~~

~~c. Examination Method and Requirements.~~

~~The examination method shall meet the intent of the requirements in ASME Section XI Appendix IV. This includes equipment, personnel, and procedure requirements, certification and calibration along with records and reports. The actual technique may be the latest industry accepted technique, provided the flaw detection capability is as good or better than the technique endorsed by the code in effect per 10 CFR 50, Section 50.55a(g). This allows the use of improvements in inspection techniques that were not included in the code in effect. However, it means that word-for-word compliance with Appendix IV of ASME Section XI may not be possible.~~

5.5 Programs and Manuals

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~d. Inspection Intervals~~

~~1. Inservice inspections shall not be more than 24 calendar months apart.~~

~~2. The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 5.5.8.d.1 are not exceeded.~~

~~3. If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals.~~

~~4. If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 5.5.8-1 requires that a third sample examination must be performed, and the results of this fall in the category C-3, the inspection frequency shall be reduced to not more than 20 month intervals. The reduction shall apply until a subsequent inspection demonstrates that a third sample examination is not required.~~

~~5. Unscheduled inspections shall be conducted in accordance with Specification 5.5.8.b on any steam generator with primary-to-secondary tube leakage exceeding Specification 3.4.13.d. All steam generators shall be inspected in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineered safeguards, or a main steam line or feedwater line break.~~

~~e. Corrective Measures~~

~~All tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired by a process such as sleeving prior to return to power from a refueling or inservice inspection condition. Sleeved tubes having sleeve degradation exceeding 40% of the nominal sleeve wall thickness shall be plugged. (Brazed joints shall not be employed.~~

~~Tubes previously subject to explosive plugging shall not be sleeved)~~

~~The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test frequencies.~~

5.5 Programs and Manuals

~~TABLE 5.5.8-1~~
~~STEAM GENERATOR TUBE INSPECTION PER UNIT~~
~~POINT BEACH UNITS 1 & 2~~

~~1ST SAMPLE EXAMINATION~~			~~2ND SAMPLE EXAMINATION~~		~~3RD SAMPLE EXAMINATION~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of S tubes per Steam Generator (S.G.)~~ ~~S=3(N/n) %~~ ~~Where:~~ ~~N is the number of steam generators in the plant = 2~~ ~~n is the number of steam generators inspected during an examination~~	~~C-1~~	~~Acceptable for continued service~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator~~	~~C-1~~	~~Acceptable for continued service~~	~~N/A~~	~~N/A~~
			~~C-2~~	~~Plug or repair tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator~~	~~C-1~~	~~Acceptable for continued service~~
					~~C-2~~	~~Plug or repair tubes exceeding plug limit. Acceptable for continued service~~
					~~C-3~~	~~Perform action required under C-3 of 1st sample examination~~
			~~C-3~~	~~Perform action required under C-3 of 1st sample examination~~	~~N/A~~	~~N/A~~
	~~C-3~~	~~Inspect essentially all tubes in this S.G., plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).~~	~~C-1 in other S.G.~~	~~Acceptable for continued service~~	~~N/A~~	~~N/A~~
			~~C-2 in other S.G.~~	~~Perform action required under C-2 of 2nd sample examination above~~	~~N/A~~	~~N/A~~
			~~C-3 in other S.G.~~	~~Inspect essentially all tubes in S.G. & plug or repair tubes exceeding the plugging limit. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).~~	~~N/A~~	~~N/A~~

5.6 Reporting Requirements

5.6.7 Tendon Surveillance Report (continued)

Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

~~(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the commission as soon as practicable.~~

5.6 Reporting Requirements

~~(b) The complete results of the steam generator tube inservice inspection shall be included in a report for the period in which the inspection was completed.~~

~~Reports shall include:~~

~~1. Number and extent of tubes inspected.~~
~~2. Location and percent of all thickness penetration for each indication.~~
~~3. Identification of tubes plugged or repaired.~~

~~(c) Reports required by Table 5.5.8-1, "Steam Generator Tube Inspection," shall provide the information required by Specification 5.6.8.(b) and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. The report shall be submitted to the Commission prior to resumption of plant operation.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 120% RTP. This is the design overpower condition for two RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops — MODES 1 and 2 satisfy Criterion 2 of the NRC Policy Statement.

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required at rated power.

In MODES 1 and 2, an OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

BASES

LCO (continued)

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; and

c. The Rod Control System is not capable of rod withdrawal, to preclude the possibility of an inadvertent control rod withdrawal and associated power excursion.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~, which has the minimum water level specified in SR 3.4.5.2. The OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One RCS loop provides sufficient circulation for these purposes. However, one additional RCS loop is required to be OPERABLE to ensure redundant capability for decay heat removal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
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

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

BASES

LCO (continued)

that are designed to validate various accident analyses values. An example of one of the tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed during the initial startup testing program, and as such should only be performed once. 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 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem 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 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 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ the Low Temperature Overpressure Protection (LTOP) enabling temperature specified in the PTLR. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. SG secondary side water temperature can be approximated by using the SG metal temperature indicator.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~, which has the minimum water level specified in SR 3.4.6.2. The OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

BASES

LCO (continued)

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 before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR. 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 4 also allows both RHR loops to be removed from operation when at least one RCS loop is in operation to allow for the performance of leakage or flow testing, as required by Technical Specifications or by regulation. This allowance is necessary based on the design of the Point Beach RHR System configuration, which requires the system to be removed from service to perform the required PIV testing.

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 (Ref. 1) when it has an adequate water level and is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes.

However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SGs is required to be ≥ 30% narrow range.

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for 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 that primary to secondary LEAKAGE from each steam generator (SG) is 500 gpd or increases to 500 gpd as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis. ~~500 gpd 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. 2) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves. The 500 gpd primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 500 gpd primary to secondary LEAKAGE is through the affected~~in one~~ generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

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.

BASES

LCO (continued)

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 Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.
~~The 500 gallons per day 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~~,~~ or 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.

BASES

ACTIONS (continued)

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or~~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 LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

The RCS water inventory balance must be met with the reactor at steady state operating conditions (i.e., stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. ~~Therefore, a~~ N~~n~~ote 1 states~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

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

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 4).

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

REFERENCES

1. FSAR Section 1.3.3.
2. FSAR, Section 14.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed 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," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.8. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on primary to secondary LEAKAGE from each SG of 500 gallons per day or is assumed to increase to 500 gallons per day as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance.

BASES

LCO (continued)

The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 500 gallons per day per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to

BASES

LCO (continued)

primary to secondary LEAKAGE induced during the accident. The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit 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 break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation

BASES

ACTIONS (continued) prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES (continued)

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

ENCLOSURE 4C

Proposed Technical Specification and Bases Pages (markup)

Prairie Island Nuclear Generating Plant Units 1 and 2

Technical Specification Pages

1.1-3	5.0-21
3.4.14-2	5.0-22
3.4.14-3	5.0-23
3.4.19-1	5.0-24
3.4.19-2	5.0-25
5.0-13	5.0-26
5.0-14	5.0-27
5.0-15	5.0-34
5.0-16	5.0-35
5.0-17	5.0-38
5.0-18	5.0-39
5.0-19	5.0-40
5.0-20	5.0-41

Bases pages

B 3.4.4-2	B 3.4.19-2
B 3.4.14-2	B 3.4.19-3
B 3.4.14-3	B 3.4.19-4
B 3.4.14-4	B 3.4.19-5
B 3.4.14-5	B 3.4.19-6
B 3.4.14-6	B 3.4.19-7
B 3.4.14-8	B 3.4.19-8
B 3.4.14-9	B 3.4.19-9
B 3.4.19-1	B 3.4.19-10

44 pages follow

1.1 Definitions (continued)

$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

LEAKAGE

LEAKAGE from the Reactor Coolant System (RCS) shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

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

3. RCS LEAKAGE through a steam generator ~~(SG)~~ to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary ~~SG~~ LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS identified LEAKAGE not within limit for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2.1 Reduce LEAKAGE to within limits.	14 hours
	OR	
	C.2.2 Be in MODE 5.	44 hours
D. Pressure boundary LEAKAGE exists. OR Primary to secondary ~~SG~~ LEAKAGE not within limit.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	---------------------------NOTES------------------------ 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operationalLEAKAGE~~leakage~~ within limits by performance of RCS water inventory balance.	24 hours
SR 3.4.14.2	--------------------------NOTE-------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify ~~steam generator tube integrity is in accordance with the Steam Generator Program~~primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG..	~~In accordance with the Steam Generator Program~~ 72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

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

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.19.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a

safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG . Leakage is not to exceed 1 gpm per SG, except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. During the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578°F).

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria:

1. Unit 1 steam generator tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

2 Unit 2 steam generator tubes that meet the following criteria shall be plugged or repaired.

~~Steam generator tubes in each unit shall be determined OPERABLE by the following:~~

~~a. Steam Generator Sample Selection and Inspection~~

~~Each steam generator shall be determined OPERABLE in accordance with the in-service inspection schedule in Specification 5.5.8.c. The in-service inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes in the single steam generator, provided the previous inspections indicated that the two steam generators are performing in a like manner.~~

~~selected for these inspections shall be selected on a random basis except:~~

1. ~~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.~~

2. ~~The first sample of tubes selected for each in-service inspection (subsequent to the preservice inspection) of each steam generator shall include:~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

(a) ~~all tubes that previously had detectable wall penetrations (> 20%) that have not been plugged or sleeve repaired in the affected area.~~

(b) ~~tubes in those areas where experience has indicated potential problems.~~

(c) ~~a tube inspection (pursuant to Specification 5.5.8.d.1.(h)) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~

3. ~~In addition to the sample required in Specification 5.5.8.b.2(a) through (c), all tubes which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the requirements of Specification 5.5.8.b.2(a).~~

4. ~~The tubes selected as the second and third samples (if required by Tables 5.5.8-1 or 5.5.8-2) during each in-service inspection may be subjected to a partial tube or sleeve inspection provided:~~

~~(a) the tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.~~

~~(b) the inspections include those portions of the tubes or sleeves where imperfections were previously found.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~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 must exhibit significant (> 10%) further wall penetrations to be included in the above percentage calculations.~~

~~5. Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.~~

~~6. 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.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~c. Inspection Frequencies~~

~~The above required in-service inspections of steam generator tubes shall be performed at the following Frequencies:~~

~~1. In-service 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.~~

~~2. If the results of the in-service inspection of a steam generator conducted in accordance with Table 5.5.8-1 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 5.5.8.c.1; the interval may then be extended to a maximum of once per 40 months.~~

~~3. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.8-1 during the shutdown subsequent to any of the following conditions:~~

~~(a) primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.14.~~

~~(b) a seismic occurrence greater than the Operating Basis Earthquake.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(c) a loss-of-coolant accident requiring actuation of the engineered safeguards.~~

~~(d) a main steam line or feedwater line break.~~

~~d. Acceptance Criteria~~

~~1. As used in this Specification:~~

~~(a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.~~

~~(b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.~~

~~(c) Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.~~

~~(d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.~~

~~(e) Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(f) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and is~~

(a) Tubes found by inservice inspection containing flaws with a depth equal to or exceeding 50% of the nominal tube wall thickness. If significant general tube thinning occurs, this ~~criteria will be~~criterion is reduced to 40% wall penetration. This criterion does not apply to tube support plate intersections to which the voltage based repair criteria apply. This criterion~~This definition~~ does not apply to the portion of the tube in the tubesheet below the F* or EF* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* or EF* distance ~~for F* or EF* tubes~~.

The F* distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet.

The EF* distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). The EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.

(b) Tubes found by inservice inspection containing flaws in~~The repair limit for~~ the pressure boundary region of any sleeve with a depth equal to or exceeing~~is~~ 25% of the nominal sleeve wall thickness. ~~This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are~~

~~being applied. Refer to Specification 5.5.8.d.4 for the repair limit applicable to these intersections.~~

(c) Tubes found by inservice inspection that are experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of tube support plates:

~~(g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.~~

~~(h) 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.~~

~~(i) Sleeving is the repair of degraded tube regions using a new Alloy 690 tubing sleeve inserted inside the parent tube and sealed at each end by welding or by replacing the lower weld in a full depth tubesheet sleeve with a hard rolled joint. The new sleeve becomes the pressure boundary spanning the original degraded tube region.~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~(j) F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet.~~

~~(k) F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.~~

~~(l) EF* Distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.~~

~~(m) EF* Tube is a tube with degradation, below the EF* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the EF* distance.~~

~~2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks or classify as F* or EF* tubes) required by Tables 5.5.8-1 and 5.5.8-2.~~

~~3. Tube repair, after April 1, 1999, using Combustion Engineering welded sleeves shall be in accordance with the methods described in the following:~~

~~CEN-629-P, Revision 03-P, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".~~

~~5.5 Programs and Manuals~~

~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~4. Tube Support Plate Repair Limit is used for the disposition of a 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 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 voltages less than or equal to 2.0 Volts will be allowed to remain in service.~~

~~(b) Steam generator tubes,~~

i ~~whose~~ with indications of potential degradation ~~is~~ attributed to predominately axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 Volts unless no degradation is detected with a rotating pancake coil (or comparable examination technique) inspection

~~, will be repaired or plugged, except as noted in Specification 5.5.8.d.4(c) below.~~

~~(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 2.0 Volts but less than or equal to the upper voltage repair limit, may remain in service if a rotating pancake coil (or comparable examination technique) inspection does not detect degradation. Steam generator tubes,~~

ii with indications of predominately axially oriented outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit. ~~will be plugged or repaired.~~

iii. ~~(d)~~ inspected during~~If~~ an unscheduled mid-cycle inspection~~ is performed~~, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.c.2.(c).i~~d.4(a), (b)~~ and 5.5.8.c.2.(c).ii above~~(c)~~. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

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

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) ~~Tube Surveillance~~ Program (continued)

CL = cycle length (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 the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.c.2.(c).i ~~d.4(a), (b)~~ and 5.5.8.c.2.(c).ii above~~(c)~~.

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

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3 and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. For Unit 1 SGs, inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. For Unit 2 SGs, inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate more than 24 effective full power months or one refueling outage (whichever is less) without being inspected. Each time a SG is inspected, all tubes within that SG which have had the F* or EF* criteria applied with be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the inspection requirements.

4. If crack indications are found in any SG tube, then the next inspection
for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

1. There are no approved SG tube repair methods for the Unit 1 SGs.

2. a. An approved SG tube repair method for the Unit 2 SGs is the use of welded sleeves in accordance with the methods described in CEN-629-P, Revision 03-P,"Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".

   b. The installation of an additional hard roll expansion greater than the F* length and below the midplane of the tubesheet allows the use of F* criteria.

   c. The installation of an additional hard roll expansion greater than the EF* length and anywhere below 2 inches from the top of the tubesheet allows the use of the EF* criteria.

5.5.9 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of the Control Room Special Ventilation System, Auxiliary Building Special Ventilation System, Shield Building Ventilation System, and the Spent Fuel Pool Special and Inservice Purge Ventilation System each operating cycle (18 months for shared systems).

Demonstrate for the Auxiliary Building Special Ventilation, Shield Building Ventilation, Control Room Special Ventilation, and Spent Fuel Pool Special and Inservice Purge Ventilation Systems that:

a. An inplace DOP test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.05% (for DOP, particles having a mean diameter of 0.7 microns);

b. A halogenated hydrocarbon test of the inplace charcoal adsorber shows a penetration and system bypass < 0.05% (for DOP, particles having a mean diameter of 0.7 microns);

~~Table 5.5.8-1~~
~~STEAM GENERATOR TUBE INSPECTION~~

~~1st SAMPLE INSPECTION~~			~~2nd SAMPLE INSPECTION~~		~~3rd SAMPLE INSPECTION~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of S Tubes per S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Repair defective tubes and inspect additional 2S tubes in this S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~C-2~~	~~Repair defective tubes and inspect additional 4S tubes in this S.G.~~	~~C-1~~	~~None~~
					~~C-2~~	~~Repair defective tubes~~
					~~C-3~~	~~Perform action for C-3 result of first sample~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~	~~N/A~~	~~N/A~~
	~~C-3~~	~~Inspect all tubes in this S.G., Repair defective tubes and inspect 2S tubes in each other S.G.~~ ~~Prompt notification to NRC~~	~~All other S.G.s are C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~Some S.G.s C-2 but no additional S.G. are C-3~~	~~Perform action for C-2 result of second sample~~	~~N/A~~	~~N/A~~
			~~Additional S.G. is C-3~~	~~Inspect all tubes in each S.G. and repair defective tubes. Prompt notification to NRC.~~	~~N/A~~	~~N/A~~

~~S=3%; When two steam generators are inspected during that outage.~~
~~S=6%; When one steam generator is inspected during that outage.~~

~~Table 5.5.8-2~~
~~STEAM GENERATOR TUBE SLEEVE INSPECTION~~

~~1st Sample Inspection~~			~~2nd Sample Inspection~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of 20% of Tube Sleeves (1)~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Inspect all remaining tube sleeves in this S.G. and plug or repair defective sleeved tubes.~~	~~C-1~~	~~None~~
			~~C-2~~	~~Plug or repair defective sleeved tubes~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~
	~~C-3~~	~~Inspect all tube sleeves in this S.G., inspect 20% of the tube sleeves in the other S.G., and plug or repair defective sleeved tubes~~	~~The other S.G. is C-1~~	~~None~~
			~~The other S.G. is C-2~~	~~Perform action for C-2 result of first sample~~
			~~The other S.G. is C-3~~	~~Inspect all tube sleeves in each S.G. and plug or repair defective sleeved tubes~~

~~(1) Each type of sleeve is considered a separate population for determination of scope expansion~~

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6.7 Steam Generator Tube Inspection Report

~~1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.~~

~~2.The results of steam generator tube in-service inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube in-service inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection, and (3) identification of tubes plugged or sleeved.~~

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

~~3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

~~4. The results of inspections performed under Specification 5.5.8.b for all tubes that have defects below the F* or EF* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:~~

~~a. Identification of F* and EF* tubes, and~~

~~b.Location and extent of degradation.~~

a. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

1. The scope of inspections performed on each SG,

2. Active degradation mechanisms found,

3. Nondestructive examination techniques utilized for each degradation mechanism,

4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

6. Total number and percentage of tubes plugged or repaired to date,

7. The results of condition monitoring, including the results of tube pulls and in-situ testing,

8. The effective plugging percentage for all plugging and tube repairs in each SG, and

9. Repair method utilized and the number of tubes repaired by each repair method.

b.~~5~~. 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 should any of the following conditions arise:

1~~a~~. 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 (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle,~~.~~

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

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

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

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

5~~e~~. 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 1E-02, notify the NRC and provide an assessment of the safety significance of the occurrence.

5.6.8 EM Report

When a report is required by Condition C or I of LCO 3.3.3, "Event Monitoring (EM) 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)	forced flow rate, which is represented by the number of RCS loops in service. Both transient and steady state analyses include the effect of flow on the departure from nucleate boiling ratio (DNBR). The transient and accident analyses for the plant have been performed assuming both RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, and rod withdrawal events (Ref. 1). The plant is designed to operate with both RCS loops in operation to maintain DNBR within limits during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant. RCS Loops - MODES 1 and 2 satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required at power. An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~.
APPLICABILITY	In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for 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 that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.~~a 1 gpm 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 USAR (Ref. 2) analysis for SGTR assumes the plant has been operating with a 5 gpm primary to secondary leak rate for a period of time sufficient to establish radionuclide equilibrium in the secondary loop. Following the tube rupture, the initial primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential when compared to the mass transfer through the ruptured tube.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes the entire 1 gpm (at 70°F) primary to secondary LEAKAGE is through the affected~~in one~~ generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

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 reactor coolant pressure boundary (RCPB). LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

Seal welds are provided at the threaded joints of all reactor vessel head penetrations (spare penetrations, full-length Control Rod Drive Mechanisms, and thermocouple columns). Although these seals are part of the RCPB as defined in 10CFR50 Section 50.2, minor leakage past the seal weld is not a fault in the RCPB or a structural integrity concern. Pressure retaining components are differentiated from leakage barriers in the ASME Boiler and Pressure Vessel Code. In all cases, the joint strength is provided by the threads of the closure joint.

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

BASES

LCO

c. Identified LEAKAGE (continued)

with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified leakage must be evaluated to assure that continued operation is safe. 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 Any One ~~Steam Generator (~~SG~~)~~

The limit of 150 gallons per day per ~~The 150 gallons per day (gpd) limit on one~~ SG is based on implementation of the Steam Generator Voltage Based Alternate Repair Criteria and ~~is more restrictive than standard operating leakage limits to provide additional margin to accommodate a crack which might grow at greater than the expected rate or unexpectedly extend outside the thickness of the tube support plate.~~the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

BASES

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 3.4.15, "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 in excess of the LCO limits must be identified or 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, B.2.1, and B.2.2

If unidentified LEAKAGE cannot be identified or 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, gaskets, and pressurizer safety valves seats is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours. If the LEAKAGE source cannot be identified within 54 hours, then the reactor must be placed in MODE 5 within

84 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

BASES

ACTIONS (continued)

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

C.1, C.2.1, and C.2.2

If RCS identified LEAKAGE, other than pressure boundary LEAKAGE~~leakage~~ or primary to secondary LEAKAGE, is not within limits, then the reactor must be placed in MODE 3 within 6 hours. In this condition, 14 hours are allowed to reduce the identified leakage to within limits. If the identified LEAKAGE is not within limits within this time, the reactor must be placed in MODE 5 within 44 hours.

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

D.1 and D.2

If RCS pressure boundary LEAKAGE exists or if primary to secondary~~SG~~ LEAKAGE (150 gpd limit) is not within limits, the reactor must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, equilibrium xenon, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. ~~Therefore, a~~ Note 1 states ~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

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

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by monitoring containment atmosphere radioactivity. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.16, "RCS Leakage Detection Instrumentation."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 24 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.14.2

~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions~~ This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.19, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 4).

BASES

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 16, issued for comment July 10, 1967, as referenced in USAR, Section 1.2.

2. USAR, Section 14.5.

3. NEI 97-06, "Steam Generator Program Guidelines."

4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.19 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed 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," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met.

BASES

BACKGROUND (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.8. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate greater than the operational LEAKAGE rate limits in LCO 3.4.14, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via atmospheric steam dumps.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. During the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted SG under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F).

BASES

APPLICABLE SAFETY ANALYSES (continued)

For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to or greater than the LCO 3.4.17, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged or repaired, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube, nor is the region of tube below the F* and EF* distances.

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the

BASES

LCO (continued)

evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity,accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation."

Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

BASES

LCO (continued)

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than an SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed those discussed in the APPLICABLE SAFETY ANALYSES section above. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.14, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to an SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to

BASES

APPLICABILITY (continued)

secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by SR 3.4.19.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if an SG tube that should have been plugged or repaired has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with an SG tube that may not have tube integrity.

BASES

ACTIONS A.1 and A.2 (continued)

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged or repaired prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS SR 3.4.19.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1 (continued)

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.19.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.19.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. The tube repair criteria delineated in Specification 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.

The Frequency of prior to entering MODE 4 following an SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged or repaired prior to subjecting the SG tubes to significant primary to secondary pressure differential.

BASES (continued)

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

ENCLOSURE 5

The following Proposed Technical Specification Pages (revised) are contained within Enclosure 5:

Enclosure 5A – Palisades Nuclear Plant

Enclosure 5B – Point Beach Nuclear Plant Units 1 and 2

Enclosure 5C – Prairie Island Nuclear Generating Plant Units 1 and 2

ENCLOSURE 5A

Proposed Technical Specification Pages (revised)

Palisades Nuclear Plant

Technical Specification Pages

1.1-4	5.0-12
3.4.13-1	5.0-13
3.4.13-2	5.0-14
3.4.17-1	5.0-15
3.4.17-2	5.0-16
5.0-11	5.0-28

12 pages follow

1.1 Definitions

LEAKAGE

a. Identified LEAKAGE (continued)

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

3. Primary Coolant System (PCS) LEAKAGE through a Steam Generator to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE (except Primary Coolant Pump seal leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an PCS component body, pipe wall, or vessel wall.

MODE

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

OPERABLE - OPERABILITY

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

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.13 PCS Operational LEAKAGE

LCO 3.4.13 PCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary leakage.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-----NOTES----- 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ---------- Verify PCS operational LEAKAGE is within limits by performance of PCS water inventory balance.	-----NOTE----- Only required to be performed during steady state operation ---------- 72 hours
SR 3.4.13.2	----- NOTE ----- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	72 hours

3.4 PRIMARY COOLANT SYSTEM (PCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.7 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 (B&PV Code) as follows:

B&PV Code terminology for inservice testing activities	Required interval for performing inservice testing activities
Weekly	≤ 7 days
Monthly	≤ 31 days
Quarterly or every 3 months	≤ 92 days
Semiannually or every 6 months	≤ 184 days
Every 9 months	≤ 276 days
Yearly or annually	≤ 366 days
Biennially or every 2 years	≤ 731 days

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

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

d. Nothing in the B&PV Code shall be construed to supersede the requirements of any Technical Specification.

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

b. Performance criteria for SG tube integrity. (continued)

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.3 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "PCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

d. Provisions for SG tube inspections. (continued)

determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

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5.5 Programs and Manuals

This page retained for page numbering

5.5 Programs and Manuals

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5.6 Reporting Requirements

5.6.6 Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring 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 to OPERABLE status.

5.6.7 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Dome Delamination tests within 90 days after completion of the tests.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

ENCLOSURE 5B

Proposed Technical Specification Pages (revised)

Point Beach Nuclear Plant Units 1 and 2

Technical Specification Pages

1.1-3	5.5-8
3.4.13-1	5.5-9
3.4.13-2	5.5-10
3.4.17-1	5.5-11
3.4.17-2	5.6-6
5.5-7	

11 pages follow

1.1 Definitions

L_a

The maximum allowable primary containment leakage rate, L_a, shall be 0.4% of primary containment air weight per day at the peak design containment pressure (P_a).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

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

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

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

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; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary 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. OR Primary to secondary LEAKAGE not within limit.	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	----------NOTES---------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ------------------------- Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	----------NOTE---------- Not required to be performed until 12 hours after establishment of steady state operation. ------------------------- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1	Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND		
	A.2	Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

    1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

    2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of the leakage rate

5.5.8 Steam Generator (SG) Program (continued)

for an individual SG. Leakage is not to exceed 500 gallons per day per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. i. Unit 1 (alloy 600 Thermally Treated tubes): Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

   ii. Unit 2 (alloy 690 Thermally Treated tubes): Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

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5.5 Programs and Manuals

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5.6 Reporting Requirements

5.6.7 Tendon Surveillance Report (continued)

Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

ENCLOSURE 5C

Proposed Technical Specification Pages (revised)

Prairie Island Nuclear Generating Plant Units 1 and 2

Technical Specification Pages

1.1-3	5.0-18
3.4.14-2	5.0-19
3.4.14-3	5.0-20
3.4.19-1	5.0-21
3.4.19-2	5.0-22
5.0-13	5.0-30
5.0-14	5.0-31
5.0-15	5.0-38
5.0-16	5.0-39
5.0-17	5.0-40

20 pages follow

1.1 Definitions (continued)

$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
LEAKAGE	LEAKAGE from the Reactor Coolant System (RCS) shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

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

3. RCS LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS identified LEAKAGE not within limit for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	C.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	C.2.1 Reduce LEAKAGE to within limits.	14 hours
	<u>OR</u>	
	C.2.2 Be in MODE 5.	44 hours
D. Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	D.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	-------------------------NOTES------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. -- Verify RCS operational LEAKAGE within limits by performance of RCS water inventory balance.	24 hours
SR 3.4.14.2	-------------------------NOTE------------------------- Not required to be performed until 12 hours after establishment of steady state operation. -- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program

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

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG inspection.	7 days
	AND	
	A.2 Plug or repair the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.19.1 Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following an SG tube inspection

5.5 Programs and Manuals (continued)

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG, except during the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria. During the implementation of steam generator repairs on Unit 2 utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578°F).

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.14, "RCS Operational Leakage".

c. Provisions for SG tube repair criteria:

1. Unit 1 steam generator tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

2. Unit 2 steam generator tubes that meet the following criteria shall be plugged or repaired.

   (a) Tubes found by inservice inspection containing flaws with a depth equal to or exceeding 50% of the nominal tube wall thickness. If significant general tube thinning occurs, this criterion is reduced to 40% wall penetration. This criterion does not apply to tube support plate intersections to which the voltage-based repair criteria apply. This criterion does not apply to the portion of the tube in the tubesheet below the F* or EF* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* or EF* distance.

   The F* distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty). The F* distance applies to roll expanded regions below the midplane of the tubesheet.

   The EF* distance is the distance from the bottom of the upper hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.67 inches (not including eddy current uncertainty). The EF* distance applies to roll expanded regions when the top of the additional roll expansion is 2.0 inches or greater down from the top of the tubesheet.

   (b) Tubes found by inservice inspection containing flaws in the pressure boundary region of any sleeve with a depth equal to or exceeding 25% of the nominal sleeve wall thickness.

   (c) Tubes found by inservice inspection that are experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates:

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

i. with indications of potential degradation attributed to predominately axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 Volts unless no degradation is detected with a rotating pancake coil (or comparable examination technique) inspection.

ii. with indications of predominately axially oriented outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit.

iii. inspected during an unscheduled mid-cycle inspection, the following mid-cycle repair limits apply instead of the limits in Specifications 5.5.8.c.2.(c).i and 5.5.8.c.2.(c).ii above. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

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

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

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

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

CL = cycle length (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 the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 5.5.8.c.2.(c).i and 5.5.8.c2.(c).ii above.

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

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. For the Unit 1 SGs, inspect 100% of the tubes at sequential periods of 144, 108, 72, and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

3. For the Unit 2 SGs, inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected. Each time a SG is inspected, all tubes within that SG which have had the F* or EF* criteria applied will be inspected in the F* and EF* regions of the roll expanded region. The region of these tubes below the F* and EF* regions may be excluded from the inspection requirements.

4. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

1. There are no approved SG tube repair methods for the Unit 1 SGs.

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

2. a. An approved SG tube repair method for the Unit 2 SGs is the use of welded sleeves in accordance with the methods described in CEN-629-P, Revision 03-P,"Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves".

   b. The installation of an additional hard roll expansion greater than the F* length and below the midplane of the tubesheet allows the use of F* criteria.

   c. The installation of an additional hard roll expansion greater than the EF* length and anywhere below 2 inches from the top of the tubesheet allows the use of the EF* criteria.

5.5 Programs and Manuals (continued)

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5.5 Programs and Manuals (continued)

This page retained for page numbering

5.5 Programs and Manuals (continued)

This page retained for page numbering

5.5 Programs and Manuals (continued)

This page retained for page numbering

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature limits and Cold Overpressure Mitigation System setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (includes any exemption granted by NRC to ASME Code Case N-514).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Changes to the curves, setpoints, or parameters in the PTLR resulting from new or additional analysis of beltline material properties shall be submitted to the NRC prior to issuance of an updated PTLR.

5.6.7 Steam Generator Tube Inspection Report

a. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

1. The scope of inspections performed on each SG,

2. Active degradation mechanisms found,

3. Nondestructive examination techniques utilized for each degradation mechanism,

4. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

5. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

6. Total number and percentage of tubes plugged or repaired to date,

7. The results of condition monitoring, including the results of tube pulls and in-situ testing,

8. The effective plugging percentage for all plugging and tube repairs in each SG, and

9. Repair method utilized and the number of tubes repaired by each repair method.

b. 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 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 (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle,

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

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

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, and

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 1E-02, notify the NRC and provide an assessment of the safety significance of the occurrence.

5.6.8 EM Report

When a report is required by Condition C or I of LCO 3.3.3, "Event Monitoring (EM) 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.