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July 13, 2005

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
License Amendment Request: Technical Specification Improvement Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process

Pursuant to 10 CFR 50.90, Calvert Cliffs Nuclear Power Plant, Inc. requests an amendment to Renewed Operating License Nos. DPR-53 and DPR-69. The proposed amendment would revise the Technical Specification requirements related to steam generator tube integrity. The change is consistent with Nuclear Regulatory Commission-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this Technical Specification improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process.

Attachment (1) provides a description of the proposed change and confirmation of applicability. Attachment (2) provides the existing TS pages marked up to show the proposed change.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of Maryland Official.

We request approval of the proposed change as soon as possible and not later than December 6, 2005 to allow for changes in steam generator tube inspection requirements during the 2006 refueling outage. These changes include addressing contractual and procedural issues and therefore, we request an implementation period of 60 days.

A001

Should you have questions regarding this matter, please contact Mr. L. S. Larragoite at (410) 495-4922.

Very truly yours,

STATE OF MARYLAND :
: TO WIT:
COUNTY OF CALVERT :

I, George Vanderheyden, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, Inc. (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of St. Mary's, this 13th day of July, 2005.

WITNESS my Hand and Notarial Seal:

Janet L. Sully
Notary Public

My Commission Expires:

March 25 2007
Date

GV/GT/bjd

Attachments: (1) Description and Assessment

(2) Proposed Technical Specification Changes

cc: P. D. Milano, NRC
S. J. Collins, NRC
Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

DESCRIPTION AND ASSESSMENT

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1.0 INTRODUCTION

The proposed license amendment revises the requirements in the Technical Specification (TS) related to steam generator tube integrity. The changes are consistent with Nuclear Regulatory Commission (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 on May 6, 2005 (70 FR 24126), as part of the consolidated line item improvement process (CLIIP).

2.0 PROPOSED CHANGE

Consistent with the NRC-approved TSTF-449, Revision 4, the proposed TS changes include:

- Revised TS 1.1, definition of LEAKAGE
- Revised TS 3.4.13, "RCS [Reactor Coolant System] Operational Leakage"

  Exception to TSTF-449, Revision 4 approved changes:

  1. The current primary to secondary operational LEAKAGE limit of 100 gallons per day (gpd)/steam generator (SG) is already lower than the TSTF approved limit of 150 gpd/SG; therefore, it has not been changed.
  2. For Surveillance Requirement (SR) 3.4.13.1, Note 1 has been added to be consistent with TSTF-449 approved Note for SR 3.4.13.2.
  3. For SR 3.4.13.2, the LEAKAGE verification is for ≤100 gpd/SG, consistent with the limit in Item 1 above.
- New TS 3.4.18, "Steam Generator (SG) Tube Integrity"
- Revised TS 5.5.9, "Steam Generator Tube Surveillance Program"
- Revised TS 5.6.9, "Steam Generator Tube Inspection Report"
- Revised Table of Content pages to reflect the proposed changes above

Proposed revisions to the TS Bases are also included in this application. Adjustments to the TSTF-449, Revision 4 approved Bases have been made to reflect the exceptions noted above. In addition, the Basis for the new TS 3.4.18 has been modified to incorporate plant specific information on steam generator tube rupture accident analysis. 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 TS 5.5.14, Technical Specification 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

Calvert Cliffs Nuclear Power Plant, Inc. 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. Calvert Cliffs Nuclear Power Plant, Inc. has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to Calvert Cliffs Units 1 and 2, and justify this amendment for the incorporation of the changes to the Calvert Cliffs TS. The only notable exception is the RCS operational primary to secondary LEAKAGE limit. The current Calvert Cliffs TS limit of 100 gpd/SG operational primary to secondary LEAKAGE is based on the current licensing basis for safety analysis assumptions approved in Reference 1. Hence, the less restrictive operational LEAKAGE limit of 150 gpd/SG approved in TSTF-449 has not been adopted.

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 following information is provided to support the NRC staff's review of this amendment application:

Plant Name, Unit No.	Calvert Cliffs Units 1 and 2
Steam Generator Model(s):	Babcock & Wilcox Replacement Steam Generators
Effective Full Power Years (EFPY) of service for currently installed SGs	Unit 1 1.76 – 2004 refueling outage Unit 2 1.82 – 2005 refueling outage
Tubing Material (e.g., 600M, 600TT, 660TT)	690TT
Number of tubes per SG	8,471
Number and percentage of tubes plugged in each SG	Unit 1 SG 11 – 0 (0.00%), SG 12 – 0 (0.00%) Unit 2 SG 21 – 3 (0.04%), SG 22 – 29 (0.34%)
Number of tubes repaired in each SG	Unit 1 SG 11 – 0, SG 12 – 0 Unit 2 SG 21 – 0, SG 22 – 0
Degradation mechanism(s) identified	1. Upper bundle fan bar support wear 2. Loose part wear
Current primary-to-secondary leakage limits:	TS Criteria at room temperature: • No pressure boundary leakage • 1 gpm unidentified leakage • 10 gpm identified leakage • 100 gpd primary to secondary leakage through any one SG
Approved Alternate Tube Repair Criteria	None
Approved SG Tube Repair Methods	None
Performance criteria for accident leakage	Primary to secondary leak rate values assumed in licensing basis accident analysis is 100 gpd per SG at room temperature conditions (Reference 1)

7.0 NO SIGNIFICANT HAZARDS EVALUATION

Calvert Cliffs Nuclear Power Plant, Inc. has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. Calvert Cliffs Nuclear

Power Plant, Inc. has concluded that the proposed determination presented in the notice is applicable to Calvert Cliffs and the determination is hereby incorporated by Reference (2) to satisfy the requirements of 10 CFR 50.91(a).

8.0 ENVIRONMENTAL EVALUATION

Calvert Cliffs Nuclear Power Plant, Inc. has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. Calvert Cliffs Nuclear Power Plant, Inc. has concluded that the staff's findings presented in that evaluation are applicable to Calvert Cliffs and the evaluation is hereby incorporated by Reference (2) for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. Calvert Cliffs Nuclear Power Plant, Inc. is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4 (except as noted in Sections 2 and 5 above), or the NRC staff's model SE published on March 2, 2005 (70 FR 10298).

10.0 REFERENCES

1. Letter from Mr. A. W. Dromerick (NRC) to Mr. C. H. Cruse (BGE), dated May 23, 1998, "Issuance of Amendments for Calvert Cliffs Nuclear Power Plant Unit No. 1 (TAC No. M97855) and Unit No. 2 (TAC No. M97856)
2. 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)

ATTACHMENT (2)

PROPOSED TECHNICAL SPECIFICATION CHANGES

Mark-up Technical Specification Pages

ii and iv

1.1-4

3.4.13-1 and 3.4.13-2

NEW 3.4-18-1 and 3.4.18-2

5.5-7 through 5.5-16

5.6-10

Bases Page ii

B 3.4.4-2

B 3.4.5-3

B 3.4.6-3

B 3.4.7-4

B 3.4.13-2 through B 3.4.13-5

NEW B 3.4.18-1 through B 3.4.18-8

1.1 Definitions

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal 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 leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

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

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolts 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

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. 100 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

Insert 3.4.13 A

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	~~Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.~~ Verify primary to secondary LEAKAGE is ≤ 100 gallons per day through any one SG.	~~In accordance with the Steam Generator Tube Surveillance Program~~ 72 hours

Insert 3.4.13 B

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 SG tube integrity shall be maintained.

<u>AND</u>

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

------------------------------------ NOTES ------------------------------------
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
	<u>AND</u>	
	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

ACTIONS

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 3.4.18.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.18.2	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

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

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

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

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

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

Insert 5.5.9

~~The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program except as specified for individual requirements. This program provides controls for the inservice inspection of steam generator tubes to ensure that structural~~

5.5 Programs and Manuals

integrity of this portion of the Reactor Coolant System is maintained. The program shall contain the requirements listed below.

a. Steam Generator Sample Selection and Inspection - The minimum number of steam generators to be inspected shall be determined as specified in Table 5.5.9-1.

b. Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Tables 5.5.9-2 and 5.5.9-3. The inservice inspection of steam generator tubes shall be performed at the Frequencies specified in Specification 5.5.9.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.d. When applying the exceptions of 5.5.9.b.1 through 5.5.9.b.3, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; 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 inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   i. All nonplugged tubes that previously had detectable wall penetrations (> 20%); and

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

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3. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the three categories specified below. In all inspections, previously degraded tubes must exhibit significant (> 10%) further wall penetrations to be included in the percentage calculations.

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.

c. <u>Inspection Frequencies</u> - The above required inservice inspections of steam generator tubes shall be performed at the following Frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power Months, but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If at least 20 percent of the

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tubes were inspected and the results were in the C-1 Category, or if at least 40 percent of the tubes were inspected and were in the C-2 Category during the previous inspection, the next inspection may be extended up to a maximum of 30 months in order to correspond with the next refueling outage if the results of the two previous inspections were not in the C-3 Category. However, if the results of either of the previous two inspections were in the C-2 Category, an engineering assessment shall be performed before operation beyond 24 months and shall provide assurance that all tubes will retain adequate structural margins against burst throughout normal operating, transient, and accident conditions until the end of the fuel cycle or 30 months, whichever occurs first. If two consecutive inspections following service under all-volatile treatment 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 inservice inspection results of a steam generator conducted in accordance with Tables 5.5.9-2 and 5.5.9-3 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.c.1; the interval may then be extended to a maximum of once per 30 or 40 months, as applicable.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Tables 5.5.9-2 and

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5.5.9-3 during the shutdown subsequent to any of the following conditions:

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

ii. A seismic occurrence greater than the Operating Basis Earthquake;

iii. A loss-of-coolant accident requiring actuation of the engineered safeguards; or

iv. A main steam line or feedwater line break.

4. The provisions of Specification SR 3.0.2 do not apply for extending the Frequency for performing inservice inspections as stated in Specifications 5.5.9.c.1 and 5.5.9.c.2.

d. <u>Acceptance Criteria</u> - As used in this Specification:

1. <u>Tubing or Tube</u> means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

2. <u>Imperfection</u> means 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. <u>Degradation</u> means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

4. <u>Degraded Tube</u> means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

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5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

7. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging, or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

i.	original tube wall	40%
ii.	ABB-Combustion Engineering leak tight sleeve wall (Unit 2 through Cycle 14 only. Not applicable for Unit 1.)	28%
iii.	ABB-Combustion Engineering Alloy 800 leak-limiting sleeve wall(Unit 2 through Cycle 14 only. Not applicable for Unit 1.)	35%

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3 above.

9. 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.

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10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

    i. ABB-Combusting Engineering Leak Tight Sleeving as described in the proprietary ABB-Combustion Engineering Report CEN-630-P, Revision 01, "Repair of 3/4" O.D. Steam Generator Tubes Using Leak Tight Sleeves," August 1996. A post-weld heat treatment during installation will be performed. (Unit 2 through Cycle 14 only. Not applicable for Unit 1.)

    ii. ABB-Combustion Engineering Alloy 800 leak-limiting sleeving as described in the Proprietary ABB Combustion Engineering Report CEN-633-P, Revision 03-P, "Steam Generator Tube Repair For Combustion Engineering Designed Plants with 3/4-.048 Inch Wall Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves, " October 1998. (Unit 2 through Cycle 14 only. Not applicable for Unit 1.)

    Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per 5.5.9.d.9 is required prior to returning previously plugged tubes to service.

e. Surveillance Completion - The Steam Generator Tube Surveillance Program is met after completing the corresponding actions (plug or repair all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Tables 5.5.9-2 and 5.5.9-3.

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Table 5.5.9-1
Minimum Number of Steam Generators to be Inspected During Inservice Inspection

Preservice Inspection	No			Yes		
No. Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

[1] The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances, the sample sequence shall be modified to inspect the most severe conditions.

[2] The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

[3] Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

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Table 5.5.9-2
Steam Generator Tube Inspection

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per steam generator	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this steam generators	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this steam generator.	C-1	None
					C-2	Plug or 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 steam generator, plug or repair defective tubes and inspect 2S tubes in each other steam generator. 24 hour verbal notification to NRC with written follow-up pursuant to Specification 5.6.9.c	All other steam generators are C-1	None	N/A	N/A
			Some steam generators C-2 but no additional steam generator are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional steam generator is C-3	Inspect all tubes in each steam generator and plug or repair defective tubes. 24 hour verbal notification to NRC with written follow-up pursuant to Specification 5.6.9.c	N/A	N/A

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

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Table 5.5.9-3
Steam Generator Repaired Tube Inspection

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A Minimum of 20% of repaired tubes[1][2]	C-1	None	N/A	N/A
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this SG.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this SG, plug defective tubes and inspect 20% of the repaired tubes in the other SG. 24-Hour verbal notification to NRC with written follow-up, pursuant to 10 CFR 50.4	Other SG is C-1	None
			Other SG is C-2	Perform action for C-2 result of first sample
			Other SG is C-3	Inspect all repaired tubes in each SG and plug defective tubes. 24-hour verbal notification to NRC with written follow-up, pursuant to 10 CFR 50.4

(1) Each repair method is considered a separate population for determination of scope expansion.
(2) The inspection of repaired tubes may be performed on tubes from either SG based on outage plans.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.9 Steam Generator Tube Inspection Report

Insert 5.6.9

a. ~~Following each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 15 days.~~

b. ~~The complete results of the steam generator tube inservice inspection during the report period shall be submitted to the NRC prior to March 1 of each year. 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; and~~

3. ~~Identification of tubes plugged or repaired.~~

c. ~~Results of steam generator tube inspections which fall into Category C-3 require verbal notification of the NRC Regional Administrator by telephone within 24 hours prior to resumption of plant operation. The written follow-up of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence and shall be submitted within the next 30 days.~~

BASES

safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are loss of coolant flow and seized rotor (Reference 1).

RCS Loops - MODEs 1 and 2 satisfy 10 CFR 50.36(c)(2)(ii), Criteria 2 and 3.

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by having both RCS loops with both RCPs 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 RCPs providing forced flow for heat transport to an SG that is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~. Steam generator, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the RPS in MODEs 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is ≥ 50 inches below normal water level* as sensed by the RPS. The minimum water level to declare the SG OPERABLE is < 50 inches below normal water level*.

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.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODEs as indicated by the LCOs for MODEs 3, 4, 5, and 6.

Operation in other MODEs is covered by: LCO 3.4.5, LCO 3.4.6, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

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* For Unit 2, the value shall remain 10 inches below the top of the feed ring through Cycle 14.

BASES

An OPERABLE loop consists of at least one OPERABLE RCP and an SG that is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~. An RCP 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 RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.6, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within a 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.

B.1

If restoration is not possible within 72 hours, the unit must be placed in MODE 4 within 12 hours. In MODE 4, the plant may be placed on the Shutdown Cooling (SDC) System. The Completion Time of 12 hours is compatible with required operation to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If no RCS loop is in operation, except as provided in Note 1 in the LCO section, all operations involving introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 must be immediately suspended. Action to restore one RCS loop to OPERABLE status and operation shall be initiated immediately and continued until one RCS loop is restored to OPERABLE

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program,~~ and has the minimum water level specified in SR 3.4.6.2.

Similarly, for the SDC System, an OPERABLE SDC loop is composed of the OPERABLE SDC pump(s) capable of providing forced flow to the SDC heat exchanger(s). Reactor coolant pumps and SDC pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs, or the SDC System.

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.5, LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

If only one required RCS loop is OPERABLE and in operation, and no SDC loops are OPERABLE, redundancy for heat removal is lost. Action must be initiated immediately to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for decay heat removal.

B.1

If one required SDC loop is OPERABLE and in operation and no RCS loops are OPERABLE, redundancy for heat removal is lost. The plant must be placed in MODE 5 within the next 24 hours. Placing the plant in MODE 5 is a conservative action with regard to decay heat removal. With only one SDC loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining SDC loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (> 200°F to < 300°F). The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 4, with only one SDC loop operating, in an orderly manner and without challenging plant systems.

BASES

An OPERABLE SDC loop 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 flow if required. ~~An OPERABLE~~ SG can perform as a heat sink 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 to remove decay heat from the core and to provide proper boron mixing. One SDC loop provides sufficient circulation for these purposes.

Operation in other MODEs is covered by: LCO 3.4.4, LCO 3.4.5, LCO 3.4.6, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1 and A.2

If the required SDC loop is inoperable and any SGs have secondary side water levels < -50 inches, redundancy for heat removal is lost. Action must be initiated immediately to restore a second SDC loop to OPERABLE status or to restore the water level in the required SGs. Either Required Action A.1 or Required Action A.2 will restore redundant decay heat removal paths. The immediate Completion Times reflect the importance of maintaining the availability of two paths for decay heat removal.

B.1 and B.2

If no SDC loop is in operation, except as permitted in Note 1, all operations involving introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 must be suspended. Action to restore one SDC loop to OPERABLE status and place it in operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of water into the RCS with a boron concentration less than that required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. When water is added without forced circulation, unmixed coolant could be introduced to the core, however water added with a boron concentration meeting

BASES

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 a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 100 gpd/SG primary to secondary LEAKAGE as the initial condition.

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

safety analysis assumption

Reference 1, Section 14.15 analysis for SGTR assumes the contaminated secondary fluid is released via the atmospheric dump valves and main steam safety valves. Most of the released radiation is due to the ruptured tube. The 100 gpd/SG primary to secondary LEAKAGE is relatively inconsequential.

The steam line break is more limiting for site radiation releases. The safety analysis for the steam line break accident assumes 100 gpd/SG primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from the steam line break accident are described in Reference 1, Section 14.14.

Reactor Coolant System operational LEAKAGE satisfies 10 CFR 50.36(c)(2)(ii), Criterion 2.

LCO

Reactor Coolant System 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

b. Unidentified LEAKAGE

One 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 the detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the Containment Structure from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled RCP seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

Insert B 3.4.13A

~~d. Primary to Secondary LEAKAGE through Any One Steam Generator~~

~~The 100 gallon per day limit on primary to secondary LEAKAGE through any one SG is consistent with SG tube sleeving commitments.~~

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.

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 four hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits

BASES

before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

or primary to secondary LEAKAGE is not within limit,

If any pressure boundary LEAKAGE exists or if unidentified, or identified ~~, or primary to secondary~~ LEAKAGE cannot be reduced to within limits within four 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 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 first appear as unidentified LEAKAGE and can only be positively identified by inspection. 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.~~

(stable temperature, power level, pressurizer and makeup tank levels, and makeup and letdown)

The RCS water inventory balance must be performed with the reactor at steady-state operating conditions ~~and near operating pressure~~.

Insert B 3.4.13B

Steady-state operation is required to perform a proper water inventory balance; calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady-state is defined as stable RCS

BASES

pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal leakoff 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. These leakage detection systems are specified in LCO 3.4.14.

Insert B3.4.13 C

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

Insert B3.4.13 D

~~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 test cannot be performed at normal operating conditions.~~

~~In the event one or more SGs are determined to not meet the requirements of the Steam Generator Tube Surveillance Program at anytime in MODEs 1 through 4, action to comply with LCO 3.0.3 must be taken.~~

REFERENCES

1. UFSAR

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

Insert B 3.4.13 E

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam Generator 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 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, LCO 3.4.5, LCO 3.4.6, and LCO 3.4.7.

Steam generator 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 5.5.9, requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, 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 5.5.9. 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 Reference 1.

APPLICABLE SAFETY ANALYSES

The 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, 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 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 100 gpd/SG or is assumed to increase to 100 gpd/SG 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 3.4.15 limits assuming the relevant Iodine spiking factors. 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 General Design Criteria (GDC) 19 (Reference 2), 10 CFR Part 100 (Reference 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 5.5.9, and described acceptable SG tube performance. 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 References 4 and 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 the total accident leakage does not exceed 100 gpd/SG.

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 and limits primary to secondary LEAKAGE through any one SG to 100 gpd. 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.

Reactor Coolant System 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 3.4.18.2. 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.

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 3.4.18.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. Reference 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.

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 technique capabilities, and inspection locations.

BASES

The Steam Generator Program defines the Frequency of SR 3.4.18.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 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 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.18.2

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 5.5.9 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 util the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessment to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency 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.

REFERENCES

1. NEI 97-06, Steam Generator Program Guidelines,
2. 10 CFR Part 50, Appendix A, GDC 19,
3. 10 CFR Part 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

INSERT 3.4.13 A

--NOTES--

1. Not required to be performed until 12 hours after establishment of steady state operation.

2. Not applicable to primary to secondary LEAKAGE.

INSERT 3.4.13 B

--NOTE--

Not required to be performed until 12 hours after establishment of steady state operation.

INSERT B 3.4.13 A

d. Primary to Secondary LEAKAGE through Any One Steam Generator

The limit of 100 gallons per day per SG is based on safety analysis assumption. This limit is more conservative than the 150 gpd/SG operational LEAKAGE performance criterion in Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines (Reference 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.

INSERT B 3.4.13 B

The surveillance is modified by two Notes. Note 1 states 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.

INSERT B 3.4.13 C

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 100 gallons per day cannot be measured accurately by an RCS water inventory balance.

INSERT B 3.4.13 D

This SR verifies that primary to secondary LEAKAGE is less or equal to 100 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 3.4.18, "Steam Generator Tube Integrity," should be evaluated. The 100 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, and makeup and letdown.

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 Electric Power Research Institute (EPRI) guidelines (Reference 4).

<u>INSERT B 3.4.13 E</u>

3. NEI 97-06, Steam Generator Program Guidelines
4. EPRI, Pressurized Water Reactor Primary-to-Secondary Leak Guidelines

<u>INSERT 5.5.9</u>

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, plugged, to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. Steam generator 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 100 gpd 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. 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. 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.

<u>INSERT 5.6.9</u>

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 5.5.9, 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,

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