Entergy

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Fred Dacimo
Site Vice President
Administration

May 31, 2006

Re: Indian Point Units 2 and 3
Dockets 50-247 and 286
NL-06-063

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

SUBJECT: License Amendment Request for Adoption of TSTF- 449 Regarding Steam Generator Tube Integrity

References: 1. Entergy Letter NL-06-028, M. Kansler to U, S, NRC, "30-Day Response to NRC Generic Letter 2006-01: Steam Generator Tube Integrity and Associated Technical Specifications", dated February 21, 2006.

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) is submitting a request for an amendment to the technical specifications (TS) for Indian Point Nuclear Generating Unit Nos. 2 & 3 (IP2 and IP3).

The proposed amendment would revise the TS requirements related to steam generator (SG) 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). This proposed amendment request also fulfills ENO's commitment provided to the NRC in response to Generic Letter 2006-01 (Reference 1).

Attachment I provides a description of the proposed change and confirmation of applicability. Attachment II provides the existing TS pages marked-up to show the proposed changes. In addition, Attachment III provides, for information purposes, the existing TS Bases pages marked-up to show the proposed changes.

Both the current IP2 TS primary-to-secondary leakage limit and the proposed limit are 150 gallons per day (gpd) through any one SG. In the background discussion supporting the CLIIP, the NRC assumed that the associated safety analyses were performed at a higher leak rate and thus margin between the analysis of record and the TS limit would exist. The IP2 safety analyses were performed using 150 gpd. Therefore, to provide additional margin as assumed in

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the CLIIP, ENO is committing (Attachment IV) to administratively maintain the IP2 primary-to-secondary leakage limit at 75 gpd through any one SG. ENO will revise the commitment when new calculations are performed which provide the margin discussed in the CLIIP. Also, wording changes were made to the Bases that may differ from the CLIIP, to reflect plant specific safety analyses. The analyses for IP3 already provide for margin to the proposed new primary-to-secondary leakage limits as intended in the CLIIP.

A copy of this application and the associated attachments are being submitted to the designated New York State official.

ENO requests approval of the proposed license amendment by January 31, 2007, with the amendment being implemented within 60 days of approval. If you should have any questions, please contact Mr. Patric W. Conroy, IPEC Licensing Manager at (914) 734-6668.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 5/31/06.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

Attachments:

I. Analysis of Proposed Technical Specification Changes Regarding Steam Generator Tube Integrity, TSTF-449.
II. Proposed Technical Specification Changes – Markup Pages
III. Proposed Technical Specification Bases Changes – Markup Pages
IV. Commitment

cc: Mr. John P. Boska, Senior Project Manager, NRC NRR DORL
Mr. Samuel J. Collins, Regional Administrator, NRC Region I
NRC Resident Inspector's Office, Indian Point 2
NRC Resident Inspector's Office, Indian Point 3
Mr. Peter R. Smith, NYSERDA
Mr. Paul Eddy, NYS Department of Public Service

ATTACHMENT I TO NL-06-063

ANALYSIS OF PROPOSED

TECHNICAL SPECIFICATION CHANGES REGARDING

STEAM GENERATOR TUBE INTEGRITY, TSTF- 449

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNITS NO. 2 and 3
DOCKET NO. 50-247 and 50-286

1.0 INTRODUCTION

The proposed license amendment revises the requirements in Technical Specification (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 on May 6, 2005 (70 FR 24126) 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:

- Revised TS definition of LEAKAGE
- Revised TS 3.4.13, "RCS [Reactor Coolant System] Operational Leakage"
- New TS 3.4.17, "Steam Generator (SG) Tube Integrity"
- Revised TS 5.5.7 (Indian Point Unit 2) and TS 5.5.8 (Indian Point Unit 3), "Steam Generator (SG) Program"
- Revised TS 5.6.7 (Indian Point Unit 2) and TS 5.6.8 (Indian Point Unit 3), "Steam Generator Tube Inspection Report"

Proposed revisions to the TS Bases are also included in this application. 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.

Both the current Indian Point Unit 2 TS primary to secondary leakage limit and the proposed limit are 150 gallons per day (gpd) through any one SG. In the background discussion supporting the CLIIP, the NRC assumed that the associated safety analyses were performed at

a higher leak rate and thus margin between the analysis of record and the TS limit would exist. The Indian Point Unit 2 safety analyses were performed using 150 gpd. Therefore, to provide additional margin as assumed in the CLIIP, ENO will administratively maintain the Indian Point Unit 2 primary to secondary leakage limit at 75 gpd through any one SG. ENO will revise the commitment when new calculations are performed which provide the margin discussed in the CLIIP. Also, wording changes were made to the Bases, that may differ from the CLIIP, to reflect plant specific safety analyses. The analyses for IP3 already provide for margin to the proposed new primary-to-secondary leakage limits as intended in the CLIIP.

5.0 TECHNICAL ANALYSIS

Entergy Nuclear Operations, Inc. (ENO) has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR10298) 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. ENO has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to Indian Point Nuclear Generating Unit Nos. 2 & 3 (Indian Point Units 2 & 3) and justify this amendment for the incorporation of the changes to the Indian Point Units 2 & 3 TS.

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 Tables are provided to support the NRC staff's review of this application for amendment.

Table One – Indian Point Unit 2

Plant Name, Unit No.	Indian Point Unit 2
Steam Generator Model(s):	Westinghouse 44F
Effective Full Power Years (EFPY) of service for currently installed SGs	5
Tubing Material	600TT
Number of tubes per SG	3214
Number and percentage of tubes plugged in each SG	21SG – 10 Tubes, 0.31% 22SG – 3 Tubes, 0.09% 23SG – 3 Tubes, 0.09% 24SG – 9 Tubes, 0.28%
Number of tubes repaired in each SG	21SG – 0 Tubes 22SG – 0 Tubes 23SG – 0 Tubes 24SG – 0 Tubes
Degradation mechanism(s) identified	WEAR (AVB)
Current primary-to-secondary leakage limits:	per SG: 150 gpd Total : 600 gpd Leakage is evaluated at what temperature condition: Room Temperature
Approved Alternate Tube Repair Criteria (ARC): 1. None	Additional information for each ARC: 1. Not Applicable
Approved SG Tube Repair Methods: 1. None	Additional information for each repair method: 1. Not Applicable
Performance criteria for accident leakage	Primary-to-secondary leak rate values assumed in licensing basis accident analysis, including assumed temperature conditions: 150 gpd per SG 600 gpd total Density of leakage assumed to be 62.4 lbm/ft3 which corresponds to a temperature of < 53°F

Table Two – Indian Point Unit 3

Plant Name, Unit No.	Indian Point Unit 3
Steam Generator Model(s):	Westinghouse 44F
Effective Full Power Years (EFPY) of service for currently installed SGs	12
Tubing Material	690TT
Number of tubes per SG	3214
Number and percentage of tubes plugged in each SG	31SG – 1 Tube, 0.03% 32SG – 6 Tubes, 0.19% 33SG – 3 Tubes, 0.09% 34SG – 4 Tubes, 0.12%
Number of tubes repaired in each SG	31SG – 0 Tubes 32SG – 0 Tubes 33SG – 0 Tubes 34SG – 0 Tubes
Degradation mechanism(s) identified	NONE (The plugged tubes, indicated above, were plugged for non-service related reasons).
Current primary-to-secondary leakage limits:	per SG: 0.3 gpm (432 gpd) Total : 1.0 gpm (1440 gpd) Leakage is evaluated at what temperature condition: Room Temperature
Approved Alternate Tube Repair Criteria (ARC): 1. None	Additional information for each ARC: 1. Not Applicable
Approved SG Tube Repair Methods 1. None	Additional information for each repair method: 1. Not Applicable
Performance criteria for accident leakage	Primary-to-secondary leak rate values assumed in licensing basis accident analysis, including assumed temperature conditions: 0.3 gpm per SG 1.0 gpm total Density of leakage assumed to be 62.4 lbm/ft3 which corresponds to a temperature of < 53°F

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

ENO has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. ENO has concluded that the proposed determination presented in the notice is applicable to Indian Point Unit Nos. 2 & 3 and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

8.0 ENVIRONMENTAL EVALUATION

ENO has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. ENO has concluded that the staff's findings presented in that evaluation are applicable to Indian Point Unit Nos. 2 & 3 and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. ENO is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298).

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)

ATTACHMENT II TO NL-06-063

MARKUP OF TECHNCIAL SPECIFICATION PAGES

FOR PROPOSED CHANGES REGARDING

STEAM GENERATOR TUBE INTEGRITY, TSTF- 449

Affected Pages (IP2)

Table of Contents ii and iv
1.1-3
3.4.13-1 and -2
New Pages 3.4.17-1 and -2
5.5-6 through 5.5-8
5.5-16 and 5.5-17
5.6-4 and 5.6.5

Affected Pages (IP3)

Table of Contents ii and iv
1.1-4
3.4.13-1 and -2
New Pages 3.4.17-1 and -2
5.0-13 through 5.0-19
5.0-36

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNITS NO. 2 and 3
DOCKET NO. 50-247 and 50-286

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3.4.8 RCS Loops - MODE 5, Loops Not Filled
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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 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, and

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.

MASTER RELAY TEST

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

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 bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

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 ~~SG~~.

steam generator (SG)

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

operational

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS 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

2. Not applicable to primary to secondary LEAKAGE

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	---------- **- NOTE** ~~2~~ 5 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. ---------- Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	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 ≤150 gallons per day through any one SG.	~~In accordance with the Steam Generator Tube Surveillance Program~~ 72 hours

Insert 3.4.13A

INSERT 3.4.13 A

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

3.4.17

3.4 REACTOR COOLANT SYSTEM (RCS)

~~3.4.20~~ Steam Generator (SG) Tube Integrity

LCO ~~3.4.20~~ 3.4.17 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
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	AND	
OR	B.2 Be in MODE 5.	36 hours
SG tube integrity not maintained.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.4.20.1~~ SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
~~SR 3.4.20.2~~ SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged ~~for repaired~~ in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

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

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

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

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

Insert 5.5.7

This program assures the continued integrity of the steam generator tubes that are a part of the primary coolant pressure boundary. Steam generator tubes shall be determined OPERABLE by the following inspection program and corrective measures:

a. Definitions

1. Imperfection is a deviation from the dimension, finish, or contour required by drawing or specification.

2. Deformation is a deviation from the initial circular cross-section of the tubing. Deformation includes the deviation from the initial circular cross-section known as denting.

3. Degradation means service-induced cracking, wastage, pitting, wear or corrosion (i.e., service-induced imperfections).

4. Degraded Tube is a tube that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

5. % Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

6. Defect is a degradation of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

7. Plugging Limit is the degradation depth at or beyond which the tube must be plugged or repaired.

8. Hot-Leg Tube Examination is an examination of the hot-leg side tube length. This shall include the length from the point of entry at the hot-leg tube sheet around the U-bend to the top support of the cold leg.

5.5 Programs and Manuals

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

9. Cold-Leg Tube Examination is an examination of the cold-leg side tube length. This shall include the tube length between the top support of the cold leg and the face of the cold-leg tube sheet.

b. Extent and Frequency of Examination

1. Steam generator examinations shall be conducted not less than 12 months nor later than twenty four calendar months after the previous examination. *

2. Scheduled examinations shall include each of the four steam generators in service.

3. Unscheduled steam generator examinations shall be required in the event there is a primary to secondary leak exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steamline or feedwater line break.

4. Unscheduled examinations may include only the steam generator(s) affected by the leak or other occurrence.

c. Basic Sample Selection and Examination

1. At least 12% of the tubes in each steam generator to be examined shall be subjected to a hot-leg examination.

2. At least 25% of the tubes inspected in Technical Specification 5.5.7.c.1 above shall be subjected to a cold-leg examination.

3. Tubes selected for examination shall include, but not be limited to, tubes in areas of the tube bundle in which degradation has been reported, either at Indian Point 2 in prior examinations, or at other utilities with similar steam generators.

4. Examination shall be by eddy current techniques as specified by the steam generator examination program submitted to the NRC in accordance with Technical Specification 5.5.7. In all cases, a probe with at least a 610-mil diameter shall be used.

* Except that the surveillance related to the steam generator tube inspection due no later than November 17, 2004, may be deferred until June 17, 2006.

5.5 Programs and Manuals

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

d. Additional Examination Criteria (Degradation)

1. If 5% or more of the tubes examined in a steam generator exhibit degradation or if any of the tubes examined in a steam generator are defective, additional examinations shall be required as specified in Table 5.5-1

2. Tubes for additional examination shall be selected from the affected area of the tube array and the examination may be limited to that region of the tube where degradation or defective tube(s) were detected.

3. The second and third sample inspections in Table 5.5-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

e. Acceptance Criteria and Corrective Action

1. Tubes shall be considered acceptable for continued service if:

   a. depth of degradation is less than 40% of the tube wall thickness, and

   b. the tube will permit passage of a 0.610" diameter probe

2. Tubes that are not considered acceptable for continued service shall be plugged or repaired.

f. Reports and Review

1. The proposed steam generator examination program shall be submitted for NRC staff review at least 60 days prior to each scheduled examination.

2. The results of each steam generator examination shall be submitted to NRC within 45 days after the completion of the examination. A significant increase in the rate of denting or significant change in steam generator condition shall be reportable immediately.

3. Restart after the scheduled steam generator examination need not be subject to NRC approval.

Table 5.5-1 (Page 1 of 2)
Steam Generator Tube Inspection

First Sample Inspection		Second Sample Inspection		Third Sample Inspection		
Minimum Size	Result	Action	Result	Action	Result	Action
12% tubes per steam generator hot leg plus 3% tubes per steam generator cold leg	C-1	- - - - - - - - - -	- - - -	- - - - - - - - - - - -	- - >	None required.
	C-2	Plug or repair defective tubes.	C-1 -	- - - - - - - - - - - -	- - >	None required.
		Inspect additional 6% tubes in this S.G.	C-2	Plug or repair defective tubes. Inspect additional 12% tubes in this S.G.	C-1->	None required.
					C-2->	Plug or repair defective tubes.
					C-3->	Go to first sample, C-3 action.
			C-3	Go to first sample. C-3 action.		
	C-3	Inspect all tubes this S.G. Plug or repair defective tubes.	All other S.G.s C-1 -	- - - - - - - - - - - -	- - >	None required.
		Inspect 6% tubes in each other S.G. if not included in the examination program.	Some S.G.s C-2 but no add'l C-3	Go to second sample. C-2 action.		
			Add'l S.G. C-3	Inspect all tubes in all S.G.s. Plug or repairs defective tubes.	- - >	Report to NRC. NRC approval req'd prior to startup.

Table 5.5-1 (Page 2 of 2)
Steam Generator Tube Inspection

Category C-1	Less than 5% of the total tubes inspected are degraded tubes and none of them is defective.
Category C-2	One or more of the total tubes inspected is defective but less than 1% of the tubes inspected are defective and less than 10% of the tubes inspected are degraded tubes.
Category C-3	More than 10% of the total inspected are degraded or more than 1% of the tubes inspected are defective.

INSERT 5.5.~~8~~ 7

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]~~ 150 gpd per SG ~~[, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program]~~.

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 ~~[or repaired]~~.

---------- Reviewer's Note ----------
Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube repair criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

[The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:

1.]

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.

---------- Reviewer's Note ----------
Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.
---

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.]

[2. 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.]

[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.

[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.

---------- Reviewer's Note ----------
Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.

1.]

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report", April 1995.

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code", August 1985;

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", August 1985; and

11. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and Cosi Condensation Model", July 1997.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided to the NRC upon issuance for each reload cycle.

5.6.6 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Steam Generator Tube Inspection Report

Insert 5.6.7

If the results of the steam generator inspection indicate greater than 1% of the inspected tubes in any steam generator exceed the repair criteria in accordance with the requirements of the Steam Generator Program, a Special Report shall be submitted within 120 days after the initial entry into MODE 4 following completion of the inspection. The report shall summarize:

a) The scope of inspections performed on each steam generator inspected in the affected unit during the current outage,

b) Active degradation mechanisms found,

c) NDE techniques utilized for each degradation mechanism

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

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

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

f) Repair method utilized and the number of tubes repaired by each repair method,

g) Total number and percentage of tubes plugged and/or repaired to date,

h) The effective plugging percentage for all plugging and tube repairs in each steam generator, and

i) The results of condition monitoring including the results of tube pulls and in-situ testing.

INSERT 5.6.~~9~~ 7

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~~ 7, 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 ~~[or repaired]~~ during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged ~~[or repaired]~~ 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 ~~[and tube repairs]~~ in each SG ~~, and]~~ .

~~[i. Repair method utilized and the number of tubes repaired by each repair method.]~~

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1.1 Definitions

LEAKAGE (continued)

system not directly connected to the atmosphere. Leakage past the pressurizer safety valve seats and leakage past the safety injection pressure isolation valves are examples of reactor coolant system leakage into closed systems.)

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 for leakage into closed systems and RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

primary to secondary

LEAKAGE (except 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 each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

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

(continued)

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. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and~~

d. ~~e.~~ ~~432~~ 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. 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

2 Not applicable to primary to secondary LEAKAGE

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	5 ----------NOTE---------- 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. ----------	-----NOTE----- Only required to be performed during steady state operation ----------
	Verify RCS Operational ~~leakage~~ 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.~~	~~In accordance with the Steam Generator Tube Surveillance Program~~

Insert 3.4.13A

Verify primary to secondary LEAKAGE is ≤150 gallons per day through any one SG.

72 hours

INSERT 3.4.13 A

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

3.4.17

3.4 REACTOR COOLANT SYSTEM (RCS)

~~3.4.20~~ Steam Generator (SG) Tube Integrity

LCO ~~3.4.20~~ 3.4.17 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
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	AND	
OR	B.2 Be in MODE 5.	36 hours
SG tube integrity not maintained.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.4.20.1~~ SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
~~SR 3.4.20.2~~ SR 3.4.17.2	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

5.5.8 Steam Generator (SG) Tube Surveillance Program

Insert 5.5.8

This program provides controls for the inservice inspection of SG tubes to assure the continued integrity of the Reactor Coolant System pressure boundary and shall include the following:

a. SG Selection and SG Tube Sample Size

The minimum sample size shall be in conformance with the requirements specified in Table 5.5-1. Selection and testing of steam generator tubes shall be made on the following basis:

1. At the first inservice inspection subsequent to the pre-service inspection, six percent of the tubes in each of two steam generators shall be inspected as a minimum.

2. At the second inservice inspection subsequent to the pre-service inspection, twelve percent of the tubes in one of the two steam generators not inspected during the first inservice inspection shall be inspected as a minimum.

3. At the third inservice inspection subsequent to the pre-service inspection, twelve percent of the tubes in the steam generator not inspected during the first two inservice inspections shall be inspected as a minimum.

4. Fourth and subsequent inservice inspections may be limited to one steam generator on a rotating schedule encompassing 12% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in like manner. 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 sequences should be modified to inspect the steam generator with the most severe conditions.

5. Unscheduled inspections should be conducted on the affected steam generator(s) in accordance with the first sample

(continued)

5.5 Programs and Manuals

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

inspection specified in Table 5.5-1 in the event of primary-to-secondary tube leaks (not including leaks originated from tube-to-tube sheet welds) exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

b. SG Tube Selection Criteria

1. Tubes for the inspection should be selected on a random basis except where experience in similar plants with similar water chemistry indicates critical areas to be inspected.

2. The first sample inspection subsequent to the pre-service inspection should include all non-plugged tubes that previously had detectable wall penetration (> 20%) and should also include tubes in those areas where experience has indicated potential problems.

3. The second and third sample inspections in Table 5.5-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

4. In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetration to be included in the percentage calculation for the result categories in Table 5.5-1.

c. Inspection FREQUENCY

1. Inservice inspections should be not less than 12 or more than 24 calendar months after the previous inspection.

(continued)

5.5 Programs and Manuals

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

2. If the results of two consecutive inspections, not including the preservice inspection, all fall into the C-1 category, the frequency of inspection may be extended to 40-month intervals. Also, if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated.

3. SR 3.0.2 is applicable to the Steam Generator Tube Surveillance Program test frequencies.

d. Classification of Test Results

1. Definitions:

Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

Degradation means a service-induced cracking, wastage, wear or corrosion.

Degraded Tube is a tube that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

% Degradation is an estimate % of the tube wall thickness affected or removed by degradation.

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

(continued)

5.5 Programs and Manuals

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

Tube Plugging Limit is the tube imperfection depth at or beyond which the tube must either be removed from service or repaired. This is considered to be an imperfection depth of 40%.

Sleeve Plugging Limit is the sleeve imperfection depth at or beyond which the sleeved tube must be removed from service or repaired. This is considered to be an imperfection depth of 40% for tube sleeves.

Tube Inspection is a full length inspection for the initial 3% sample specified in Table 5.5-1. Supplemental sample inspections (after the initial 3% sample) may be limited to a partial length inspection concentrating on those locations where degradation has been found.

2. Results Classifications

The results of each sampling examination of a steam generator shall be classified into the following three categories:

Category C-1: Less than 5% of the total tubes inspected are degraded tubes and none are defective.

Category C-2: One or more but not more than 1% of the total tubes inspected are defective or between 5 and 10% of the tubes inspected are degraded tubes.

Category C-3: More than 10% of the total tubes inspected are degraded or more than 1% of the tubes inspected are defective.

e. Corrective Action

1. The inspection result classification and the corresponding required action are specified in Table 5.5-1.

(continued)

5.5 Programs and Manuals

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

2. All leaking tubes and defective tubes should be plugged or repaired.

3. Results of steam generator tube inspections which fall into Category C-3 of Table 5.5-1 require notification of the NRC within 15 days of this determination.

4. NRC approval prior to startup is required when SG Tube Inspections identify Category C-3 degradation or defects in more than one SG.

(continued)

5.5 Programs and Manuals

TABLE 5.5-1 (page 1 of 2)
STEAM GENERATOR TUBE INSPECTION

First Sample		Second Sample		Third Sample	
Result	Required Action	Result	Required Action	Result	Required Action
C-1	Acceptable for Service	C-1	Acceptable for Service	C-1	Acceptable for Service
C-2	Plug or Repair defective tubes AND Inspect additional 2S tubes in this SG	C-1	Acceptable for Service	N/A	N/A
		C-2	Plug or Repair defective tubes AND Inspect additional 4S tubes in this SG	C-1	Acceptable for Service
				C-2	Plug or Repair defective tubes AND Acceptable for Service
				C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG
		C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG	N/A	N/A

(continued)

5.5 Programs and Manuals

TABLE 5.5-1 (page 2 of 2)
STEAM GENERATOR TUBE INSPECTION

First Sample		Second Sample		Third Sample	
Result	Required Action	Result	Required Action	Result	Required Action
C-3	Inspect all tubes in this SG AND Plug or Repair defective tubes AND Inspect 2S tubes in each other SG	All other SGs C-1	Acceptable for Service	N/A	
		Some SGs C-2 AND No other SG C-3	Plug or Repair defective tubes AND Inspect additional 4S tubes in this SG		
		Other SG C-3	Inspect all tubes in all SGs. AND Plug or Repair defective tubes AND Report and NRC Approval required prior to startup		

Sample Size shall consist of a minimum of S tubes per Steam Generator (SG)

$S=3(N/n)\%$

where:

N is the number of steam generators in the plant

n is the number of steam generators inspected during an examination

Result Classifications (C-1, C-2 and C-3) are defined in Section 5.5.8.d.

~~(continued)~~

INSERT 5.5.~~9~~ 8

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 for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program]~~.

0.3 gpm per SG not to exceed 1 gpm total leakage.

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 ~~[or repaired]~~.

---------- Reviewer's Note ----------
Alternate tube repair criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube repair criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

[The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:

1.]

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.

---------- Reviewer's Note ----------
Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.
---

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.]

[2. 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.]

[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.

[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.

---------- Reviewer's Note ----------
Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.

1.]

5.6 Reporting Requirements

5.6.7 Post Accident Monitoring Instrumentation (PAM) Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the next 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Steam Generator Tube Inspection Report

Insert 5.6.8 →

~~The number of tubes plugged or repaired in each steam generator during each inservice inspection of steam generator tubes shall be reported to the Commission within 15 days following the inspection.~~

~~Complete results of the steam generator tube inservice inspections shall be reported in writing on an annual basis for the period in which the inspection was completed per Specification 5.5.8. This report shall include:~~

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

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

~~c. Identification of the tubes plugged and the tubes repaired.~~

INSERT 5.6.~~9~~ 8

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~~ 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 ~~[or repaired]~~ during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged ~~[or repaired]~~ 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 ~~[and tube repairs]~~ in each SG. ~~and~~

~~[i. Repair method utilized and the number of tubes repaired by each repair method.]~~

ATTACHMENT III TO NL-06-063

MARKUP OF TECHNCIAL SPECIFICATION BASES PAGES

FOR PROPOSED CHANGES REGARDING

STEAM GENERATOR TUBE INTEGRITY, TSTF- 449
(for information only)

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ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNITS NO. 2 and 3
DOCKET NO. 50-247 and 50-286

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming four 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 four 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 four RCS loop operation. For four RCS loop operation, the steady state DNB analysis generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) by assuming a maximum for the power level. This is the design overpower condition for four RCS loop operation. The allowable value for the nuclear overpower (high flux) trip in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," is based on this analysis assumption and bounds 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 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, four pumps are required at rated 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 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)

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

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation. This Note does not prohibit injection into the RCS of water with a boron concentration that is equal to or greater than the minimum boron concentration needed to meet the SDM requirement in LCO 3.1.1 even if the water being injected has a lower boron concentration than the water already in the RCS.

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.

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. 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. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

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,"

BASES

LCO (continued)

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.

An OPERABLE RHR loop consists of one OPERABLE RHR pump and one OPERABLE RHR heat exchanger as well as associated piping and valves to transfer heat between the RHR heat exchanger and the core. Although either RHR heat exchanger may be credited for either RHR loop, one RHR heat exchanger must be OPERABLE for each OPERABLE RHR loop.

RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2,"
LCO 3.4.5,	"RCS Loops - MODE 3,"
LCO 3.4.7,	"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 loop is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

BASES

LCO (continued)

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.

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

If the SGs being credited as the redundant method of decay heat removal depend on natural circulation (Ref. 1) for decay heat removal, the SGs are considered OPERABLE only if:

a. RCS loop and reactor pressure vessel filling and venting was completed during startup or pressurizer level has been maintained ≥ 10% to assure no draining of SG tubes,

b. The ability to pressurize and control pressure in the RCS to ≥ 100 psig is maintained to control the potential for flashing and void formation at the top of the SG tubes; and,

c. The Condensate Storage Tank (CST) and the motor driven auxiliary feedwater pump(s) associated with the credited SGs is operable.

Additionally, the level indication in the pressurizer should be monitored since an unexplained rise in level could be indicative of non-condensable gases coming out of solution.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be ≥ 0% narrow range.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2;"
LCO 3.4.5, "RCS Loops - MODE 3;"

BASES

BACKGROUND (continued)

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

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 because 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 600 gpd primary to secondary LEAKAGE (150 gpd in each of the four SGs) (Ref. 3).

Insert B3.4.13A

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from locked rotor and steam line break (SLB) accidents. 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.

primary to secondary LEAKAGE

In the analyses of the locked rotor and steamline break accidents the 150 gpd per steam generator leakage is the primary contributor to the doses. In the locked rotor accident analysis the leakage is the only path by which the contaminated primary coolant can be transferred out of the reactor coolant system. Activity transferred to the secondary side during a locked rotor event is subject to partitioning as steam is released to the atmosphere, but the leakage is the main source of the secondary activity. In the steamline break analysis all activity transferred to the faulted steam generator by the leakage is released without partitioning. Activity transferred to the intact steam generators is subject to partitioning so this leakage is relatively inconsequential. In the SGTR analysis the ruptured tube is the primary path by which the contaminated primary coolant is transferred out of the reactor coolant system. Activity transferred to the intact steam generators is much lower due to the lower leak rate and is subject to partitioning so this leakage is relatively inconsequential. The dose consequences resulting from these events are within the applicable limits (Ref. 3). The dose acceptance limits are defined by Regulatory Guide 1.183 (Ref. 4).

The safety analysis for the SLB accident assumes 150 gpd primary to secondary LEAKAGE through the affected generator as an initial condition.

the 450 gpd primary to secondary LEAKAGE safety analysis assumption

A limit of 150 gpd of primary to secondary leakage per SG is established as part of the performance criteria for the SG tube surveillance program recommended in Reference 5. Monitoring and limiting primary to secondary leakage is an important defense in depth measure for monitoring overall tube integrity during operation. SG leakage monitoring and the

BASES

LCO (continued)

Insert B 3.4.13B →

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

~~The 150 gpd limit for primary to secondary LEAKAGE in each of the four SGs is part of the performance criteria for the SG tube surveillance program required by Technical Specification 5.5.7, Steam Generator (SG) Tube Surveillance Program, and Reference 5. SG leakage monitoring and the associated limit allows operators to safely respond to situations in which tube integrity becomes impaired before significant leakage or tube failure occurs.~~

~~Note that primary to secondary LEAKAGE is also counted as identified LEAKAGE in accordance with Technical Specification 1.1, "Definitions."~~

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

or primary to secondary LEAKAGE is not within limit,

If any pressure boundary LEAKAGE exists, or if unidentified ~~LEAKAGE,~~ or 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 Surveillance is modified by two Notes

The RCS water inventory balance must be met with the reactor at steady state operating conditions. ~~Therefore, a~~ Note ~~is added allowing~~ 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.

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.1 (continued)

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."

Insert B 3.4.13C

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 B 3.4.13D

~~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. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. WCAP-16157, "Indian Point Unit 2 Stretch Power Uprating Licensing Report," January 2004.
4. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

Insert B 3.4.13E

~~5. NEI 97-06, Steam Generator Program Guidelines, Rev. 1.~~

600 gallons per day

INSERT B 3.4.13 A

600 gallons per day

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~~ equal to the conditions assumed in the safety analysis.

INSERT B 3.4.13 B

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. 4). 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 C

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.

~~INSERT B 3.4.13 D (BWOG)~~

~~This SR verifies that primary to secondary LEAKAGE is less than 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 5. 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. 5).~~

INSERT B 3.4.13 D ~~(WOG)~~

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 ~~3.4.20,~~ 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 5. 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. 5).

~~INSERT B 3.4.13 D (CEOG)~~

~~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 3.4.18, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. 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. 5).~~

INSERT B 3.4.13 E

5. → ~~4.~~ NEI 97-06, "Steam Generator Program Guidelines."

6. → ~~5.~~ EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

~~B 3.4.20~~ Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

B 3.4.17

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.

5.5.7

Specification ~~5.5.9,~~ "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification ~~5.5.9,~~ 5.5.7 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.~~ 5.5.7 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 ~~only briefly~~ released to the atmosphere via safety valves ~~and the majority is discharged to the main condenser~~ or atmospheric relief 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 ~~1 gallon per minute~~ 600 gallons per day or is assumed to increase to ~~1 gallon per minute~~ 600 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 ~~[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, a 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.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification ~~5.5.9,~~ 5.5.7 "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.

BASES

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 ~~[1 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage.]~~ 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.

150 gallons per day per SG.

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, "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 ~~[or repaired]~~ in accordance with the Steam Generator Program as required by ~~[SR 3.4.20.2]~~ 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 ~~[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.

BASES

Actions (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 ~~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 3.4.20.1~~ 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.

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 3.4.20.1.~~ 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 ~~5.5.9~~ 5.5.7 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

~~SR 3.4.20.2~~ SR 3.4.17.2

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 ~~5.5.9~~ 5.5.7 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 a 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.

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."

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(continued)

BASES (continued)

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, four pumps are required at rated 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 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.

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, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops – MODE 3";
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 the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

(continued)

BASES

LCO (continued)

therefore, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure redundant decay heat removal capability.

The Note permits all RCPs to be not be in operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit performance of required tests or maintenance that can only be performed with all reactor coolant pumps not in operation. The 1 hour time period specified is acceptable because operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

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

a. No operations are permitted that would dilute the RCS boron concentration, thereby 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.

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. 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. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with RTBs in the closed position. The least stringent

(continued)

BASES

LCO (continued)

Note 1 permits all RCPs and RHR pumps to not be in operation for # 1 hour per 8 hour period. The purpose of the Note is to permit performance of required tests or maintenance that can only be performed with no forced circulation. The 1 hour time period is acceptable because 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 test or maintenance 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 reactor coolant pump starting requirements of LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), must be met before the start of an RCP with any RCS cold leg temperature less than or equal to the LTOP arming temperature. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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.

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.

(continued)

BASES

LCO (continued)

Note 3 requires that the reactor coolant pump starting requirements of LCO 3.4.12, Low Temperature Overpressure Protection (LTOP), must be met before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature less than the LTOP arming temperature specified in LCO 3.4.12, Low Temperature Overpressure Protection (LTOP). 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.

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 with forced flow or natural circulation 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 two SGs is required to be ≥ 71% wide range.

Loops filled is based on the ability to use the SGs as a backup means of decay heat removal. The RCS loops are considered filled provided that pressurizer level has been maintained D10%. The loops are also considered filled following the completion of filling and venting the RCS. The ability to pressurize the RCS to ≥100 psig and to control pressure must be established to take credit for use of the SGs as backup decay heat removal. This is to prevent flashing and void formation at the top of the SG tubes

(continued)

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 events resulting in steam discharge to the atmosphere assumes ~~a range of primary to secondary LEAKAGE from 0.1 gpm to 10 gpm as the initial condition.~~

Insert B 3.4.13A

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 released via safety valves and atmospheric dump valves. The ~~1~~ 0.9 gpm primary to secondary LEAKAGE is relatively inconsequential.

safety analysis assumption

The SLB is more limiting for site radiation 432 gpd releases. The safety analysis for the SLB accident assumes ~~a range of~~ primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50.67 and 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.

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

(continued)

BASES

LCO (continued)

in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount and is consistent with the capability of the equipment required by LCO 3.4.15, RCS Leakage Detection Instrumentation. 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, the leakage into closed systems or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

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

~~Total primary to secondary LEAKAGE amounting to 1 gpm (1440 gpd) through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

(continued)

BASES

LCO (continued)

Insert B 3.4.13B

~~e. Primary to Secondary LEAKAGE through Any One SG~~

~~The 432 gallons per day (0.3 gpm) 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.

Leakage past PIVs or other leakage into closed systems is that leakage that can be accounted for and contained by a system not directly connected to the atmosphere. Leakage past PIVs or other leakage into closed systems is not included in the limits for either identified or unidentified LEAKAGE but PIV leakage must be within the limits specified for PIVs in LCO 3.4.14, "RCS Pressure Isolation Valves (PIV)." Leakage past PIVs or other leakage into closed systems is quantified before being exempted from the limits for 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.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified ~~LEAKAGE,~~ or primary to secondary LEAKAGE is not within limit, or identified LEAKAGE, ~~or primary to secondary LEAKAGE~~ cannot be

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 blowdown systems.~~

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

The surveillance is modified by two Notes.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

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

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

Insert B 3.4.13C

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

Insert B 3.4.13D (WOG)

~~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. 10 CFR 50, Appendix A, GDC 30.

2. FSAR, Section 14.

Insert B 3.4.13E

INSERT B 3.4.13 A

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.

INSERT B 3.4.13 B

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. 4). 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 C

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.

INSERT B 3.4.13 D (BWOG)

This SR verifies that primary to secondary LEAKAGE is less than 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 5. 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. 5).

INSERT B 3.4.13 D ~~(WOG)~~

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 ~~3.4.20,~~ 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 5. 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. 5).

~~INSERT B 3.4.13 D (CEOG)~~

~~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 3.4.18, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. 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. 5).~~

INSERT B 3.4.13 E

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

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

~~B 3.4.20~~ Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

B 3.4.17

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.

5.5.8

Specification ~~5.5.9~~ "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification ~~5.5.9~~ 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 ~~5.5.9~~ 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 ~~only briefly~~ released to the atmosphere via safety valves ~~and the majority is discharged to the main condenser.~~ SG or atmospheric relief 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 ~~[1~~ gallon per minute~~]~~ or is assumed to increase to ~~[1~~ gallon per minute~~]~~ 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 ~~[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, a 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.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, ~~5.5.9,~~ "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.

BASES

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 ~~[1 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage.]~~ 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.

0.3 gpm per SG not to exceed 1 gpm total leakage.

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, "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 ~~[or repaired]~~ in accordance with the Steam Generator Program as required by SR 3.4.20.2. 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 ~~[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.

BASES

Actions (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 ~~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 3.4.20.1~~ ← 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.

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 3.4.20.1.~~ 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 ~~5.5.9~~ 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 3.4.20.2~~ SR 3.4.17.2

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 ~~5.5.9~~ 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 a 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.

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."

ATTACHMENT IV TO NL-06-063

ENTERGY COMMITMENT

FOR PROPOSED CHANGES REGARDING

STEAM GENERATOR TUBE INTEGRITY, TSTF- 449

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNITS NO. 2 and 3
DOCKET NO. 50-247 and 50-286

COMMITMENT REGARDING PROPOSED LICENSE AMENDMENT REQUEST REGARDING STEAM GENERATOR TUBE INTEGRITY

ID	Commitment Description	Due Date
NL-06-063-A	ENO will administratively maintain the Indian Point Unit 2 TS primary-to-secondary leakage limit at 75 gpd through any one Steam Generator.	Upon implementation of approved license amendment