April 14, 2005 TSTF-05-05

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

SUBJECT: TSTF-449, Revision 4, "Steam Generator Tube Integrity"

Dear Sir or Madam:

Enclosed for NRC consideration is Technical Specification Task Force Traveler TSTF-449, Revision 4, "Steam Generator Tube Integrity." This revision incorporates minor changes made to TSTF-449 in response to comments on the NRC's draft Safety Evaluation, published in the Federal Register on March 2, 2005.

We request that NRC review of the Traveler continue to be granted a fee waiver pursuant to the provisions of 10 CFR 170.11. Specifically, the request is to support NRC generic regulatory improvements (steam generator tube integrity), in accordance with 10 CFR 170.11(a)(1)(iii). This request is consistent with the NRC letter to A. R. Pietrangelo on this subject dated January 10, 2003.

Should you have any questions, please do not hesitate to contact us.

Bert Yates for

Wesley Sparkman (WOG) Michael Crowthers (BWROG)

Brian Woods (WOG) Paul Infanger (BWOG)

Enclosure

G014

cc: J. Riley, NEI
Thomas H. Boyce, Technical Specifications Section, NRC

Add: A Pietrangelo
T. Boyce

11921 Rockville Pike, Suite 100, Rockville, MD 20852
Phone: 301-984-4400, Fax: 301-984-7600
Email: tstf@excelservices.com
Administered by EXCEL Services Corporation

The B&W Owners Group WOG BWR OWNERS' GROUP

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Steam Generator Tube Integrity

NUREGs Affected: ☑ 1430 ☑ 1431 ☑ 1432 ☐ 1433 ☐ 1434

Classification: 1) Technical Change Recommended for CLIIP?: Yes

Correction or Improvement: Improvement NRC Fee Status: Exempt

Benefit: Reduces Testing

Industry Contact: Wes Sparkman, (205) 992-5061, wasparkm@southernco.com

See attached justification.

Revision History

OG Revision 0 Revision Status: Closed

Revision Proposed by: NEI SG Task Force

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 10-Feb-03

Owners Group Comments:
Distributed to WOG core group and NEI Steam Generator Task Force.

Owners Group Resolution: Approved Date: 03-Mar-03

TSTF Review Information

TSTF Received Date: 04-Mar-03 Date Distributed for Review: 04-Mar-03

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 12-Mar-03

NRC Review Information

NRC Received Date: 17-Mar-03

NRC Comments:
Revised based on RAIs received on Catawba lead plant submittal.

Final Resolution: NRC Requests Changes: TSTF Will Revise

TSTF Revision 1 Revision Status: Closed

TSTF Revision 1 Revision Status: Closed

Revision Proposed by: NEI SG Task Force

Revision Description:
Catawba Nuclear Station is the lead plant for the changes in the Traveler. The Catawba submittal was made on February 25, 2003. The NRC responded with Requests for Additional Information (RAIs) on April 30, May 29, and July 21, 2003. A revised Catawba submittal was transmitted on June 9, 2003 and a further revision was submitted on July 30, 2003. This revision of TSTF-449 incorporates the applicable changes made in the Catawba submittal as a result of the RAIs.

The specific changes are:

1) The Bases of the RCS loop specifications (3.4.4, 3.4.6, and 3.4.7) are revised to not state that a SG is OPERABLE "in accordance with the Steam Generator Tube Surveillance Program." In order for a SG to be OPERABLE, it must be capable of performing its safety function. SG tube integrity (required for RCS boundary integrity) is one important aspect of SG OPERABILITY, but is not the only aspect. Primary and secondary side water level, the ability to pressurize the system, instrumentation and controls and other design and performance requirements are also necessary. Therefore, the Bases are revised to state that the SG must be OPERABLE. The requirements for OPERABILITY are not explicitly described and are left to the definition of OPERABILITY. This is consistent with the treatment of Reactor Coolant Pumps in the same sentence.

2) All references to repairing SG tubes which satisfy the tube repair criteria are bracketed. If tube repair methods have been approved for a particular plant, the material in the brackets will be retained. For those plants without approved tube repair methods, the material is not incorporated in the plant-specific Specifications. In order to maintain consistent numbering in the 5.5.9 program, the optional section describing approved tube repair methods is moved to the end of the program.

3) SR 3.4.13.2 is revised. The Frequency is changed from "In accordance with the SG Program" to "72 hours." A Note is added which states, "Not required to be performed until 12 hours after establishment of steady state operation." This change is made in response to an NRC RAI stating that 10 CFR 50.36 requires the SR Frequency to be stated in the Technical Specifications. The Bases are revised to reflect the changes to the Specification.

4) The Steam Generator Tube Integrity Specification is revised based on the NRC RAIs. Required Action A.1 is revised to state, "Verify tube integrity of the affected tube(s) is maintained until the next inspection."

5) Specification 5.6.9 is revised to allow 180 days instead of 120 days to provide the report and to eliminate the threshold for submitting the report. The order of the required information is changed to list the plant-specific (e.g., bracketed) items last in order to maximize consistency.

6) Specification 5.5.9, Steam Generator Program, is revised in response to NRC RAIs:

a) The structural integrity performance criteria was revised to be consistent with the definition used in the Catawba submittal. For discussion of the differences between the Revision 0 and Revision 1 Traveler, see the Catawba RAI responses.

b) Paragraph c is revised to eliminate the phrase "prior to entry into MODE 4." This information is already contained in the referencing Specification.

c) The requirements for SG tube inspections is replaced in its entirety.

TSTF Review Information

TSTF Revision 1 Revision Status: Closed

TSTF Received Date: 03-Jun-03 Date Distributed for Review: 27-Aug-93

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

TSTF to revise based on results of SGTF / NRC discussions on Structural Integrity criterion.

TSTF Resolution: Approved Date: 09-Sep-03

NRC Review Information

NRC Received Date: 10-Sep-03

NRC Comments:

The NEI Steam Generator Task Force and the NRC agreed to changes to the Structural Integrity Performance Criteria and to include a definition of "collapse."

Final Resolution: Superceded by Revision

TSTF Revision 2 Revision Status: Closed

Revision Proposed by: NEI SG Task Force

Revision Description:
The NEI Steam Generator Task Force and the NRC worked to resolve issues regarding the Structural Integrity Performance Criteria (SIPC). The proposed Steam Generator Program in the TS Administrative Controls has been revised to reflect the new definition. In addition, the LCO Bases of the Steam Generator Tube Integrity Specification have been revised to incorporate a definition of "collapse" based on an NRC recommendation. Also a definition of what constitutes "significant" loads and a discussion of the treatment of thermal loads was added to the LCO Bases. The justification has been revised to discuss these changes.

The references are revised to eliminate the specific EPRI technical report numbers. These numbers change with each revision, but the titles remain the same. Therefore, the document titles are the appropriate reference to the latest version of the documents.

TSTF Review Information

TSTF Received Date: 10-Sep-04 Date Distributed for Review: 10-Sep-04

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 07-Oct-04

NRC Review Information

NRC Received Date: 07-Oct-04

NRC Comments:

The NRC and the NEI SG Task Force negotiated additional changes to TSTF-449. A revision will be submitted.

Final Resolution: NRC Requests Changes: TSTF Will Revise

TSTF Revision 2 **Revision Status: Closed**

TSTF Revision 3 **Revision Status: Closed**

Revision Proposed by: TSTF

Revision Description:
The definition of the accident induced leakage limit is revised to clarify the connection between the accident induced leakage limit performance criterion, approved alternate repair criteria that change the licensing basis, and specific leakage limits. The Reviewer's Note to the provisions for SG tube repair criteria in the TS Steam Generator Program was revised to state that any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair methods should be listed in the TS Program.

The Steam Generator Tube Integrity Bases references are revised to eliminate the specific EPRI technical report numbers. These numbers change with each revision, but the titles remain the same. Therefore, the document titles are the appropriate reference to the latest version of the documents.

The justification was revised to eliminate references to preparing a degradation assessment prior to an outage or SG inspection. Emergent conditions may result in an outage or SG inspection being started before a degradation assessment is prepared.

The Bases of the RCS Operational LEAKAGE specification, Applicable Safety Analyses section, is revised to clarify that the LCO limits LEAKAGE to 1 gpm through all SGs, but some accident analyses assume that the entire 1 gpm is through one steam generator in order to maximize the consequences of the accident.

The Bases of the RCS Operational LEAKAGE, the primary to secondary LEAKAGE Surveillance, are revised. The Bases previously discussed performance of the SR when primary system radioactivity is below the minimum detectable amount. This discussion has been eliminated and the Bases revised to state that the sampling is determined in accordance with the EPRI guidelines.

The justification, Section 13, "Reporting Requirements," was revised to clarify 10 CFR 50.72 and 10 CFR 50.73 reporting requirements.

An editorial change is made to the definition of LEAKAGE. The definition uses the term "SG LEAKAGE" but that term is never used in the Technical Specifications or Bases. The term used in the Technical Specifications and Bases is "primary to secondary LEAKAGE." Therefore, the definition is revised to use the term "primary to secondary LEAKAGE" instead of "SG LEAKAGE."

Various editorial corrections were made.

The marked ISTS pages were remarked on Revision 3 of the ISTS NUREGs.

TSTF Review Information

TSTF Received Date: 04-Jan-05 Date Distributed for Review: 04-Jan-05

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 14-Jan-05

TSTF Revision 3 Revision Status: Closed

NRC Review Information

NRC Received Date: 14-Jan-05

NRC Comments:

As of 1/28/05, the CLIIP is in concurrence.

Final Resolution: NRC Releases for FRN Final Resolution Date: 28-Jan-05

TSTF Revision 4 Revision Status: Active

Revision Proposed by: TSTF

Revision Description:
TSTF-449, Revision 3 is revised to address comments made on the NRC's draft Safety Evaluation, published in the Federal Register on March 2, 2005.

The following changes are made:

1) Required Action A.1 of the Steam Generator Tube Integrity Specification is revised from "Verify tube integrity of the affected tube(s) is maintained until the next inspection" to "Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection." This change is appropriate because Required Action A.2 requires the affected tube(s) to be plugged or repaired "Prior to entering MODE 4 following the next refueling outage or SG tube inspection." A corresponding change is made to the Bases of Required Action A.1.

2) On Page 3 of the justification, the title given for Reference 4 is corrected and made consistent with the title given in the Reference section.

3) The Bases of the second Surveillance of the new Steam Generator Tube Integrity specification is revised to eliminate Reference 7. There is no reference 7 and the existing Reference 1 is sufficent.

4) The LCO Bases of the new Steam Generator Tube Integrity specification is corrected to use the word "significant" instead of "significantly" to be consistent with the definition in the Bases and in the justification. In the NUREG-1430 Bases, this occurs on page B 3.4.17-3, second paragraph, line 13. The same change is made to the NUREG-1431 and NUREG-1432 Bases.

TSTF Review Information

TSTF Received Date: 11-Apr-05 Date Distributed for Review: 11-Apr-05

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 14-Apr-05

NRC Review Information

NRC Received Date: 15-Apr-05

TSTF Revision 4 **Revision Status: Active**

Affected Technical Specifications

1.1	Definitions Change Description: Revised definition of LEAKAGE	
LCO 3.4.5 Bases	RCS Loops - MODE 3	NUREG(s)- 1430 1431 1432 Only
LCO 3.4.6 Bases	RCS Loops - MODE 4	NUREG(s)- 1430 1431 1432 Only
LCO 3.4.7 Bases	RCS Loops - MODE 5, Loops Filled	NUREG(s)- 1430 1431 1432 Only
Bkgnd 3.4.13 Bases	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
S/A 3.4.13 Bases	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
LCO 3.4.13	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
LCO 3.4.13 Bases	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
Ref. 3.4.13 Bases	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
Action 3.4.13.A	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
Action 3.4.13.A Bases	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
Action 3.4.13.B	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
Action 3.4.13.B Bases	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
SR 3.4.13.1	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
SR 3.4.13.1 Bases	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
SR 3.4.13.2	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
SR 3.4.13.2 Bases	RCS Operational LEAKAGE	NUREG(s)- 1430 1431 1432 Only
5.5.9	Steam Generator Tube Surveillance Program Change Description: Renamed Steam Generator Program	NUREG(s)- 1430 1431 1432 Only
5.6.9	Steam Generator Tube Inspection Report	NUREG(s)- 1430 1431 1432 Only
3.4.17	SG Tube Integrity Change Description: New specification	NUREG(s)- 1430 Only

3.4.17 Bases	SG Tube Integrity		NUREG(s)- 1430 Only
	Change Description:	New specification	
LCO 3.4.4 Bases	RCS Loops - MODES 1 and 2		NUREG(s)- 1431 1432 Only
3.4.18	SG Tube Integrity		NUREG(s)- 1431 Only
	Change Description:	New specification	
3.4.18 Bases	SG Tube Integrity		NUREG(s)- 1431 Only
	Change Description:	New specification	
3.4.20	SG Tube Integrity		NUREG(s)- 1432 Only
	Change Description:	New specification	
3.4.20 Bases	SG Tube Integrity		NUREG(s)- 1432 Only
	Change Description:	New specification	

1.0 DESCRIPTION

The proposed change revises the Improved Standard Technical Specification (ISTS), NUREGs 1430, 1431, and 1432, Specification 3.4.13, "RCS Operational LEAKAGE," Specification 5.5.9, "Steam Generator Tube Surveillance Program, and Specification 5.6.9, "Steam Generator Tube Inspection Report," and adds a new specification for Steam Generator Tube Integrity. The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines," (Reference 1).

2.0 PROPOSED CHANGE

The proposed change will:

- Revise Technical Specification 3.4.13, "RCS Operational LEAKAGE"

  The proposed change revises TS 3.4.13, "RCS Operational LEAKAGE," by reducing the allowable leakage from any one steam generator (SG) to 150 gallons per day. The proposed change also deletes the existing LCO 3.4.13.d since it is enveloped by the revised LCO and revises the Conditions and Surveillances to clarify the requirements related to primary to secondary LEAKAGE.

  SR 3.4.13.2 is changed from verifying SG tube integrity to requiring verification that primary to secondary LEAKAGE is within limit. SG tube integrity is verified under a new LCO. A new Note is added to SR 3.4.13.1 to indicate that this surveillance in not applicable to primary to secondary LEAKAGE. A Note is added to SR 3.4.13.2 stating that the SR is not required to be performed until 12 hours after establishment of stable plant conditions. This is consistent with the existing Note on SR 3.4.13.1.

  TS Bases changes are made to reflect the changes proposed to the Technical Specifications.

- Revise Technical Specification 5.5.9, "Steam Generator Tube Surveillance Program"

  The proposed change revises TS 5.5.9, "Steam Generator Tube Surveillance Program," to delete the Reviewer's Note and to require a Steam Generator Program to be established and implemented to ensure that SG tube integrity is maintained, and to describe SG condition monitoring, performance criteria, repair methods, repair criteria, and inspection intervals. The title of TS 5.5.9 is revised from Steam Generator Tube Surveillance Program to Steam Generator Program.

- Revise Technical Specification 5.6.9, "Steam Generator Tube Inspection Report"

  The proposed change to TS 5.6.9, "Steam Generator Tube Inspection Report," deletes the Reviewer's Note and provides the requirements for and contents of the SG tube inspection report. The reporting requirements are revised to require a report within 180 days of initial entry into MODE 4 following a steam generator inspection.

- Add a Steam Generator Tube Integrity Specification

  The proposed change adds a new Technical Specification entitled "Steam Generator Tube Integrity," and associated Bases. The proposed Specification requires that SG tube integrity be maintained and requires that all SG tubes that satisfy the repair criteria be plugged or repaired in accordance with the Steam Generator Program.

- Revise the TS Bases for Specifications 3.4.4, 3.4.5, 3.4.6, and 3.4.7

  The TS Bases for NUREG-1431 and NUREG-1432, Specification 3.4.4, "RCS Loops – MODES 1 and 2," and NUREG-1430, -1431, and 1432, Specification 3.4.5, "RCS Loops – MODE 3," 3.4.6, "RCS Loops – MODE 4," and 3.4.7, "RCS Loops – MODE 5, Loops Filled," are revised to eliminate the reference to the Steam Generator Tube Surveillance Program as the method of establishing Steam Generator OPERABILITY.

- Revise the TS definition of "LEAKAGE"

  An editorial change is made to the definition of LEAKAGE. The definition uses the term "SG LEAKAGE" but that term is never used in the Technical Specifications or Bases. The term used in the Technical Specifications and Bases is "primary to secondary LEAKAGE." Therefore, the definition is revised to use the term "primary to secondary LEAKAGE" instead of "SG LEAKAGE."

3.0 BACKGROUND

The SG tubes in pressurized water reactors 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 upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they act as a heat transfer surface between the primary and secondary systems to remove heat from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system.

Steam generator tube integrity is necessary in order to satisfy the tubing's safety functions. Maintaining tube integrity ensures that the tubes are capable of performing their intended safety functions consistent with the plant licensing basis, including applicable regulatory requirements.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. Steam generator tubes have experienced 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. When the degradation of the tube wall reaches a prescribed repair criterion, the tube is considered defective and corrective action is taken.

The criteria governing structural integrity of SG tubes were developed in the 1970s and assumed uniform tube wall thinning. This led to the establishment of a through wall SG tube repair criteria (e.g. 40 percent) that has historically been incorporated into most pressurized water reactor (PWR) Technical Specifications and has been applied, in the absence of other repair criteria, to all forms of SG tube degradation where sizing techniques are available. Since the basis of the through wall depth criterion was 360° wastage, it is generally considered to be conservative for other mechanisms of SG tube degradation. The repair criterion does not allow licensees the flexibility to manage different types of SG tube degradation. Licensees must either use the through wall criterion for all forms of degradation or obtain approval for use of more appropriate repair criteria that consider the structural integrity implications of the given mechanism.

For the last several years, the industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP), has developed a generic approach to improving SG performance referred to as "Steam Generator Degradation Specific Management" (SGDSM). Under this approach, different methods of inspection and different repair criteria may be developed for different types of degradation. A degradation specific approach to managing SG tube integrity has several important benefits. These include:

- improved scope and methods for SG inspection,
- industry incentive to continue to improve inspection methods, and
- development of plugging and repair criteria based on appropriate NDE parameters.

As a result, the assurance of SG tube integrity is improved and unnecessary conservatism is eliminated.

Over the course of this effort, the SGMP has developed a series of EPRI guidelines that define the elements of a successful SG Program. These guidelines include:

- "Steam Generator Examination Guideline" (Reference 2),
- "Steam Generator Integrity Assessment Guideline" (Reference 3),
- "Steam Generator In-situ Pressure Test Guideline" (Reference 4),
- "PWR Primary-to-Secondary Leak Guideline" (Reference 5),
- "Primary Water Chemistry Guideline" (Reference 6), and
- "Secondary Water Chemistry Guideline" (Reference 7).

These EPRI Guidelines, along with NEI 97-06 (Ref. 1), tie the entire Steam Generator Program together, while defining a comprehensive, performance based approach to managing SG performance.

In parallel with the industry efforts, the NRC pursued resolution of SG performance issues. In December of 1998, the NRC Staff acknowledged that the Steam Generator Program described by NEI 97-06 (Ref. 1) and its referenced EPRI Guidelines provides an acceptable starting point to use in the resolution of differences between it and the staff's proposed Generic Letter and draft Regulatory Guide (DG-1074). Since then the industry and the NRC have participated in a series of meetings to resolve the differences and develop the regulatory framework necessary to implement a comprehensive Steam Generator Program.

Revising the existing regulatory framework to accommodate degradation specific management is the most appropriate way to address the issues of regulatory stability, resource expenditure, use of state-of-the-art inservice inspection techniques, repair criteria, and enforceability. The NRC Staff has stated that an integrated approach for addressing SG tube integrity is essential and that materials, systems, and radiological issues that pertain to tube integrity need to be considered in the development of the new regulatory framework.

4.0 TECHNICAL ANALYSIS

The proposed changes do not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The primary coolant activity limit and its assumptions are not affected by the proposed changes to the standard technical specifications. The proposed changes are an improvement to the existing SG inspection requirements and provide additional assurance that the plant licensing basis will be maintained between SG inspections.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube.

For design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor, the SG tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. For accidents that do not involve fuel damage, the reactor coolant activity levels are at the technical specification values. For accidents that do involve fuel damage, the primary coolant activity values are a function of the amount of activity released from the damaged fuel.

The consequences of these design basis accidents are, in part, functions of the radioactivity levels in the primary coolant and the accident primary to secondary LEAKAGE rates. As a result, limits are included in the plant technical specifications for operational LEAKAGE and for DOSE EQUIVALENT I-131 in primary coolant to ensure the plant is operated within its analyzed condition.

The proposed technical specification change includes a reduction in the current technical specification RCS operational LEAKAGE limit. The limit of 150 gallons per day of primary to secondary LEAKAGE through any one SG is based on operating experience as an indication of one or more tube leaks. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

The other technical specification changes proposed are in general a significant improvement over current requirements. They replace an outdated prescriptive technical specification with one that references Steam Generator Program requirements that incorporate the latest knowledge of SG tube degradation morphologies and the techniques developed to manage them.

The requirements being proposed are more effective in detecting SG degradation and prescribing corrective actions than those required by current technical specifications. As a result, these proposed changes will result in added assurance of the function and integrity of SG tubes.

The table below and associated sections describe in detail and provide the technical justification for the proposed changes.

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Operational primary to secondary LEAKAGE	$\leq$1 gpm total through all SGs and $\leq$ [720] or [500] gallons per day through any one SG	RCS Oper. LEAKAGE TS $\leq$ 150 gallons per day through any one SG	1
RCS primary to secondary LEAKAGE through any one SG not within limits	Reduce LEAKAGE to within limits in 4 hours or be in MODE 3 in 6 hours and in MODE 5 in 36 hours	RCS Oper. LEAKAGE TS - Be in MODE 3 in 6 hours and in MODE 5 in 36 hours	2
RCS LEAKAGE determined by water inventory balance (SR 3.4.13.1)	Note states: Not required to be performed until 12 hours after establishment of steady state operation	Added new Note indicating SR not applicable to primary to secondary LEAKAGE.	3
SG Tube integrity verification (SR 3.4.13.2)	Verify in accordance with the SG Tube Surveillance Program	RCS Oper. LEAKAGE TS: Revised the SR to verify primary to secondary LEAKAGE every 72 hours. Added Note stating "Not required to be performed until 12 hours after establishment of steady state operation."	4
Frequency of verification of tube integrity	6 to 40 months depending on SG category defined by previous inspection results.	SG Tube Integrity TS – Requires Surveillance Frequency in accordance with TS 5.5.9, Steam Generator Program. Frequency is dependent on tubing material and the previous inspection results and the anticipated defect growth rate. Steam Generator Program – Establishes maximum inspection intervals	5

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Tube sample selection	Based on SG Category, industry experience, random selection, existing indications, and results of the initial sample set - 3% times the number of SGs at the plant as a minimum	Steam Generator Program and implementing procedures - Dependent on a pre-outage evaluation of actual degradation locations and mechanisms, and operating experience – 20% of all tubes as a minimum.	6
Inspection techniques	Not specified	SG Tube Integrity TS – SR 3.4.20.1 requires that tube integrity be verified in accordance with the Steam Generator Program. Steam Generator Program and implementing procedures – Establishes requirements for qualifying NDE techniques. Requires use of qualified techniques in SG inspections. Requires a pre-outage evaluation of potential tube degradation morphologies and locations and an identification of NDE techniques capable of finding the degradation.	7
Inspection Scope	Hot leg point of entry to (typically) the first support plate on the cold leg side of the U-bend	Steam Generator Program procedures – Inspection scope is defined by the degradation assessment that considers existing and potential degradation morphologies and locations. Explicitly requires consideration of entire length of tube from tube-sheet weld to tube-sheet weld.	8

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Performance criteria	Operational LEAKAGE ≤ 1 gpm total or ≤ [720] or [500] gallons per day through any one SG. No criteria specified for structural integrity or accident induced leakage.	RCS Oper. LEAKAGE TS – Operational leakage ≤ 150 gallons per day through any one SG. SG Tube Integrity TS – Requires that tube integrity be maintained. TS 5.5.9 – Defines structural integrity and accident induced leakage performance criteria which are dependent on design basis limits. Provides provisions for condition monitoring assessment to verify compliance.	9
Repair criteria	Plug or repair tubes with imperfections extending [>40%] through wall and alternate criteria approved by NRC and through wall depth based criteria for repair techniques approved by the NRC. Approved alternate repair criteria listed in the Technical Specification.	TS 5.5.9 –Criteria unchanged	10
ACTIONS	Performance Criteria not defined. Primary to secondary LEAKAGE limit and actions included in the Tech Specs. Plug or repair tubes exceeding repair criteria.	RCS Oper. LEAKAGE TS and SG Tube Integrity TS – Contains primary to secondary LEAKAGE limit, SG tube integrity requirements and ACTIONS required upon failure to meet performance criteria. Plug or repair tubes satisfying repair criteria.	11

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Repair methods	Methods (except plugging) require previous approval by the NRC. Approved methods listed in Technical Specification.	TS 5.5.9 –Requirements unchanged	12
Reporting requirements	[Plugging and repair report required 15 days after each inservice inspection, 12 month report documenting inspection results, and reports in accordance with §50.72 when the inspection results fall into category C-3.]	CFR - Serious SG tube degradation (i.e., tubing fails to meet the structural integrity or accident induced leakage criteria) requires reporting in accordance with 50.72 or 50.73. TS 5.5.9 - 180 days after the initial entry into MODE 4 after performing a SG inspection	13
Definitions SG Terminology	Normal TS definitions (i.e., Definitions Section) did not address SG Program issues. The Definitions Section uses the term "SG LEAKAGE."	TS 5.5.9, TS Bases, Steam Generator Program procedures – Includes Steam Generator Program terminology applicable only to SGs. The Definitions Section is revised to use the term "primary to secondary LEAKAGE."	14

Section 1: Operational LEAKAGE

The primary to secondary LEAKAGE limit has been reduced to ≤150 gallons per day through any one SG. 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. This together with the allowable accident induced leakage limit helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR 100 and GDC 19 dose limits or other NRC approved licensing basis for postulated faulted events.

This limit also contributes to meeting the GDC 14 requirement that the reactor coolant pressure boundary "have an extremely low probability of abnormal leakage, of rapidly propagating to failure, and of gross rupture." The proposed Surveillance references the Steam Generator Program. The Steam Generator Program uses the EPRI Primary-to-Secondary Leak Guideline (Ref. 5) to establish sampling requirements for determining primary to secondary LEAKAGE and plant shutdown requirements if leakage limits are exceeded. The guidelines ensure leakage is effectively monitored and timely action is taken before a leaking tube exceeds the performance criteria. The Frequency for determining primary to secondary LEAKAGE is unchanged (i.e., 72 hours and within 12 hours after establishing stable operating conditions).

The proposed revision to the technical specification requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly more conservative than the existing technical specification limit of [1 gpm] total primary to secondary LEAKAGE through all SGs that is based on an initial condition of the safety analysis.

Section 2: Operational LEAKAGE Actions

If primary to secondary LEAKAGE exceeds 150 gallons per day through any one SG, a plant shutdown must be commenced. MODE 3 must be achieved in 6 hours and MODE 5 in 36 hours. The existing technical specifications allow 4 hours to reduce primary to secondary LEAKAGE to less than the limit. The proposed technical specification removes this allowance.

The removal of the 4 hour period during which primary to secondary LEAKAGE can be reduced to avoid a plant shutdown results in a technical specification that is significantly more conservative than the existing RCS Operational LEAKAGE specification. This change is consistent with the Steam Generator Program that also does not allow 4 hours before commencing a plant shutdown.

Section 3: RCS Operational LEAKAGE Determined by Water Inventory Balance

The proposed change adds a second Note to SR 3.4.13.1 that makes the water inventory balance method not applicable to determining primary to secondary LEAKAGE. This change is proposed because primary to secondary LEAKAGE as low as 150 gallons per day through any one SG cannot be measured accurately by an RCS water inventory balance. This change is necessary to make the surveillance requirement appropriate for the proposed LCO.

Section 4: SG Tube Integrity Verification

The current SR 3.4.13.2 requires verification of tube integrity in accordance with the SG Tube Surveillance Program. This surveillance is no longer appropriate since tube integrity is addressed through the addition of a new SG Tube Integrity Specification. Specification 3.4.13 now applies specifically to primary to secondary LEAKAGE. Surveillance Requirement 3.4.13.2 has been changed to verify the LCO requirement on primary to secondary LEAKAGE only. Steam generator tube integrity is verified in accordance with a SR in the SG Tube Integrity Specification.

The Steam Generator Program and the EPRI "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines" (Ref. 5) provide guidance on leak rate monitoring. During normal operation the program depends upon continuous process radiation monitors and/or radiochemical grab sampling in accordance with the EPRI guidelines. The monitoring and sampling frequency increases as the amount of detected LEAKAGE increases or if there are no continuous radiation monitors available.

The Surveillance Frequency is unchanged. Determination of the primary to secondary LEAKAGE is required every 72 hours. The SR is modified by a Note stating the SR is not required to be performed until 12 hours after establishment of stable operating conditions. As stated above, additional monitoring of primary to secondary LEAKAGE is also required by the Steam Generator Program based upon guidance provided in Reference 5.

Section 5: Frequency of Verification of SG Tube Integrity

The current technical specifications contain prescriptive inspection intervals which depend on the condition of the tubes as determined by the last SG inspection. The tube condition is classified into one of three categories based on the number of tubes found degraded and defective. The minimum inspection interval is no less than 12 and no more than 24 months unless the results of two consecutive inspections are in the best category (no additional degradation), and then the interval can be extended to 40 months.

The surveillance Frequency in the proposed Steam Generator Tube Integrity specification is governed by the requirements in the Steam Generator Program and specifically by References 2 and 3. The proposed Frequency is also prescriptive, but has a stronger engineering basis than the existing technical specification requirements. The interval is dependent on tubing material and whether any active degradation is found. The interval is limited by existing and potential degradation mechanisms and their anticipated growth rate. In addition, a maximum inspection interval is established in Specification 5.5.9.

The maximum inspection interval requirement for Alloy 600 mill annealed tubing (600MA) is "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." This Frequency is at least as conservative as the current technical specification requirement.

The maximum inspection interval for Alloy 600 thermally treated tubing is "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."

The maximum inspection interval for Alloy 690 thermally treated tubing is "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." Even though the maximum interval for Alloy 600 thermally treated tubing and Allow 690 thermally treated tubing is slightly longer than allowed by current technical specifications, it is only applicable to SGs with advanced materials, it is only achievable early in SG life and only if the SGs are free from active degradation. In addition, the interval must be supported by an evaluation that shows that the performance criteria will continue to be met at the next SG inspection. Taken in total, the proposed inspection intervals provide a larger margin of safety than the current requirements because they are based on an engineering evaluation of the tubing condition and potential degradation mechanisms and growth rates, not only on the previous inspection results. As an added safety measure, the Steam Generator Program requires a minimum sample size at each inspection that is significantly larger than that required by current technical specifications (20 percent versus 3 percent times the number of SGs in the plant); thus providing added assurance that any degradation within the SGs will be detected and accounted for in establishing the inspection interval.

The proposed maximum inspection intervals are based on the historical performance of advanced SG tubing materials. Reference 8 shows that the performance of Alloy 600TT and 690TT is significantly better than the performance of 600MA tubing, the material used in SG tubing at the time that the current technical specifications were written. There have been very few instances of cracking in 600TT tubes in a U.S. SG and this degradation appears to be limited to a small number of tubes in specific SGs that were left with high residual stress as a result of a problem in their manufacturing process. The mechanism is not a result of operational degradation. There are no known instances of cracking in 690TT tubes in either the U.S. or international SGs.

In summary, the proposed change is an improvement over the current technical specification. The current technical specification bases inspection intervals on the results of previous inspections; it does not require an evaluation of expected performance. The proposed technical specification uses information from previous plant inspections as well as industry experience to evaluate the length of time that the SGs can be operated and still provide reasonable assurance that the performance criteria will be met at the next inspection. The actual interval is the shorter of the evaluation results and the requirements in Reference 3. Allowing plants to use the proposed inspection intervals maximizes the potential that plants will use improved techniques and knowledge since better knowledge of SG conditions supports longer intervals.

Section 6: SG Tube Sample Selection

The current technical specifications base tube selection on SG conditions and industry and plant experience. The minimum sample size is 3% of the tubes times the number of SGs in the plant. The proposed change refers to the Steam Generator Program degradation assessment guidance for sampling requirements. The minimum sample size is 20% of the tubes inspected.

The Steam Generator Program requires the preparation of a degradation assessment. The degradation assessment is the key document used for planning a SG inspection, where inspection plans and related actions are determined, documented, and communicated. The degradation assessment addresses the various reactor coolant pressure boundary components within the SG (e.g., plugs, sleeves, tubes, and components that support the pressure boundary.) In a degradation assessment, tube sample selection is performance based and is dependent upon actual SG conditions and plant operational experience and of the industry in general. Existing and potential degradation mechanisms and their locations are evaluated to determine which tubes will be inspected. Tube sample selection is adjusted to minimize the possibility that tube integrity might degrade during an operating cycle beyond the limits defined by the performance criteria. The EPRI Steam Generator Examination Guidelines (Ref. 2) and EPRI Steam Generator Integrity Assessment Guidelines (Ref. 3) provide guidance on degradation assessment.

In general, the sample selection considerations required by the current technical specifications and the requirements in the Steam Generator Program as proposed by this change are consistent, but the Steam Generator Program provides more guidance on selection methodologies and incorporation of industry experience and requires more extensive documentation of the results. Therefore the sample selection method proposed by this change is more conservative than the current technical specification requirements. In addition, the minimum sample size in the proposed requirements is larger.

Section 7: SG Inspection Techniques

The Surveillance Requirements proposed in the Steam Generator Tube Integrity specification require that tube integrity be verified in accordance with the requirements of the Steam Generator Program. The Steam Generator Program uses the EPRI Steam Generator Examination Guidelines (Ref. 2) to establish requirements for qualifying NDE techniques and maintains a list of qualified techniques and their capabilities.

The Steam Generator Program requires the performance of a degradation assessment and refers utilities to EPRI Steam Generator Examination Guidelines (Ref. 2) and EPRI Steam Generator Integrity Assessment Guidelines (Ref. 3) for guidance on its performance. The degradation assessment will identify current and potential new degradation locations and mechanisms and NDE techniques that are effective in detecting their existence. Tube inspection techniques are chosen to reliably detect flaws that might progress during an operating cycle beyond the limits defined by the performance criteria.

The current technical specifications contain no requirements on NDE inspection techniques. The proposed change is an improvement over the current technical specifications that contained no similar requirement.

Section 8: SG Inspection Scope

The current technical specifications include a definition of inspection that specifies the end points of the eddy current examination of each tube. Typically an inspection is required from the point of entry of the tube on the hot leg side to some point on the cold leg side of the tube, usually at the first tube support plate after the U-bend. This definition is overly prescriptive and simplistic and has led to interpretation questions in the past.

The Steam Generator Program states, "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." The Steam Generator Program provides extensive guidance and a defined process, the degradation assessment, for determining the extent of a tube inspection. This guidance takes into account industry and plant specific history to determine potential degradation mechanisms and the location that they might occur within the SG. This information is used to define a performance based inspection scope targeted on plant specific conditions and SG design.

The proposed change is an improvement over the current technical specifications because it focuses the inspection effort on the areas of concern, thereby minimizing the unnecessary data that the NDE analyst must review to identify indication of tube degradation.

Section 9: SG Performance Criteria

The proposed change adds a performance-based Steam Generator Program to the Technical Specifications. A performance-based approach has the following attributes:

- measurable parameters,
- objective criteria to assess performance based on risk-insights,
- deterministic analysis and/or performance history, and
- licensee flexibility to determine how to meet established performance criteria.

The performance criteria used for SGs are based on tube structural integrity, accident induced leakage, and operational LEAKAGE. The structural integrity and accident induced leakage criteria were developed deterministically and are consistent with the plant's licensing basis. The operational LEAKAGE criterion was based on providing an effective measure for minimizing the frequency of tube ruptures at normal operating and faulted conditions. The proposed structural integrity and accident induced leakage performance criteria are new requirements. The performance criteria are specified in Specification 5.5.9. The requirements and methodologies established to meet the performance criteria are documented in the Steam Generator Program. The current technical specifications contain only the operational LEAKAGE criterion; therefore the proposed change is more conservative than the current requirements.

The SG performance criteria identify the standards against which performance is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle.

The structural integrity performance criterion is:

> "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.

The structural integrity performance criterion is based on providing reasonable assurance that a SG tube will not burst during normal operation or postulated accident conditions.

Adjustments to include contributing loads are addressed in the applicable EPRI guidelines.

Normal steady state full power operation is defined as the conditions existing during MODE 1 operation at the maximum steady state reactor power as defined in the design or equipment specification. Changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or T_{hot} should be assessed and included if significant.

The definition of normal steady state full power operation is important as it relates to application of the safety factor of three in the structural integrity performance criterion. The criterion requires "...retaining a safety factor of 3.0 under normal steady state full power operation primary to secondary pressure differential...". The application of the safety factor of three to normal steady state full power operation is founded on past NRC positions, accepted industry practice, and the intent of the ASME Code for original design and evaluation of inservice components. The assumption of normal steady state full power operating pressure differential has been consistently used in the analysis, testing and verification of tubes with stress corrosion cracking for verifying a safety factor of three against burst. Additionally, the 3ΔP criterion is measurable through the condition monitoring process.

The actual operational parameters may differ between cycles. As a result of changes to these parameters, reaching the differential pressure in the equipment specification may not be possible during plant operations. Evaluating to the pressure in the design or equipment specification in these cases would be an unnecessary conservatism. Therefore, the definition allows adjustment of the 3ΔP limit for changes in these parameters when necessary. Further guidance on this adjustment is provided in Appendix M of the EPRI Steam Generator Integrity Assessment Guidelines (Ref. 3).

The accident induced leakage performance criterion is:

> "The primary to secondary accident induced leakage rate for all design basis accidents, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. 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]."

Primary to secondary LEAKAGE is a factor in the activity releases outside containment resulting from a limiting design basis accident. The potential dose consequences from primary to secondary LEAKAGE during postulated design basis accidents must not exceed the radiological limits imposed by 10 CFR Part 100 guidelines, or the radiological limits to control room personnel imposed by GDC-19, or other NRC approved licensing basis.

In most cases when calculating offsite doses, the safety analysis for the limiting Design Basis Accident, other than a steam generator tube rupture, assumes a total of 1 gpm primary to secondary LEAKAGE as an initial condition. Revision 3 of the Standard Technical Specifications limited the amount of RCS Operational LEAKAGE to 1 gpm from all SGs, with 500 or 720 gallons per day from the worst generator, since the initial safety analyses assumed that leakage under accident conditions would not exceed the limit on Operational LEAKAGE. More recent experience with degradation mechanisms involving tube cracking has revealed that leakage under accident conditions can exceed the level of operating leakage by orders of magnitude. The NRC has concluded (Item Number 3.4 in Attachment 1 to Reference 14) that additional research is needed to develop an adequate methodology for fully predicting the effects of leakage on the outcome of some accident sequences. Therefore, a separate performance criterion was established for accident induced leakage. The limit for accident induced leakage is 1 gpm or the plant's design basis, whichever is less, unless a greater leakage rate has been approved as part of an alternate repair criteria. Use of an increased accident induced leakage limit approved in conjunction with an alternate repair criteria is limited to the specific criteria and type of degradation for which it was granted and is described in the SG Program.

The operational LEAKAGE performance criterion is:

> "The RCS operational primary to secondary LEAKAGE through any one steam generator shall be limited to 150 gallons per day."

Plant shutdown will commence if primary to secondary LEAKAGE exceeds 150 gallons per day at room temperature conditions from any one SG.

The operational LEAKAGE performance criterion is documented in the Steam Generator Program and implemented in Specification 3.4.13, "RCS Operational LEAKAGE."

Proposed Administrative Specification 5.5.9 contains the performance criteria and is more conservative than the current technical specifications. The current technical specifications do not address the structural integrity and accident induced leakage criteria. In addition, the primary to secondary LEAKAGE limit (150 gallons per day per SG) included in the proposed changes to Technical Specification 3.4.13, "RCS Operational LEAKAGE," is more conservative than the primary to secondary LEAKAGE limit in the current RCS operational LEAKAGE specification.

Section 10: SG Repair Criteria

Repair criteria are those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging.

Tube repair criteria are established for each active degradation mechanism. Tube repair criteria are either the standard through-wall depth-based criterion (e.g., 40% through-wall for most plants) or through-wall depth based criteria for repair techniques approved by the NRC, or other Alternate Repair Criteria (ARC) approved by the NRC such as a voltage-based repair limit per Generic Letter 95-05 (Ref. 12). A SG degradation-specific management strategy is followed to develop and implement an ARC.

The surveillance requirements of the proposed Steam Generator Integrity specification require that tubes that satisfy the tube repair criteria be plugged or repaired in accordance with approved methods. SG tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged-on-detection" and their integrity should be assessed. It cannot be guaranteed that every flaw will be detected with a given eddy current technique and, therefore, it is possible that some flaws will not be detected during an inspection. If a flaw is discovered and it is determined that this flaw would have satisfied the repair criteria at the time of the last inspection of the affected tube, this does not mean that the Steam Generator Program was violated. However, it may be an indication of a shortcoming in the inspection program.

Any plant-specific alternate repair criteria approved for a licensee are listed in Technical Specification 5.5.9. These are the same criteria that are listed in the existing Technical Specifications. In addition, Technical Specification 5.5.9 lists any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

Section 11: ACTIONS

The RCS Operational LEAKAGE and Steam Generator Tube Integrity specifications require the licensee to monitor SG performance against performance criteria in accordance with the Steam Generator Program.

During plant operation, monitoring is performed using the operational LEAKAGE criterion. Exceeding that criterion will lead to a plant shutdown in accordance with Technical Specification 3.4.13. Once shutdown, the Steam Generator Program will ensure that the cause of the operational LEAKAGE is determined and corrective actions are taken to prevent recurrence. Operation may resume when the requirements of the Steam Generator Program have been met. This requirement is unchanged from the current technical specifications.

Also during plant operation the licensee may discover an error or omission that indicates a failure to implement a required plugging or repair during a previous SG inspection. Under these circumstances, the licensee is expected to take the actions required by Condition A in the Steam Generator Tube Integrity specification. If a performance criterion has been exceeded, a principal safety barrier has been challenged and 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) require NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence. The Steam Generator Program additionally requires that the report contain information on the performance criteria exceeded and the basis for the planned operating cycle. The current technical specifications only address operational LEAKAGE during operations and therefore do not include the proposed requirement.

During MODES 5 and 6, the operational LEAKAGE criterion is not applicable, and the SGs will be inspected as required by the surveillance in the Steam Generator Tube Integrity specification. A condition monitoring assessment of the "as found" condition of the SG tubes will be performed to determine the condition of the SGs with respect to the structural integrity and accident leakage performance criteria. If the performance criteria are not met, the Steam Generator Program requires ascertaining the cause and determining corrective actions to prevent recurrence. Operation may resume when the requirements of the Steam Generator Program have been met.

The proposed technical specification's change to the ACTIONS required upon exceeding the operational leakage criterion is conservative with respect to the current technical specifications as explained in Section 2 above.

The current technical specifications do not address ACTIONS required while operating if it is discovered that the structural integrity or accident induced leakage performance criteria or a repair criterion are exceeded, so the proposed change is conservative with respect to the current technical specifications.

If performance or repair criteria are exceeded while shutdown, the affected tubes must be repaired or plugged. A report will be submitted to the NRC in accordance with Technical Specification 5.6.9. The changes in the required reports are discussed in Section 13 below.

<u>Section 12: SG Repair Methods</u>

Repair methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a SG tube is not a repair.

The purpose of a repair is typically to reestablish or replace the RCPB. The proposed Steam Generator Tube Integrity surveillance requirements requires that tubes that satisfy the tube repair criteria be plugged or repaired in accordance with the Steam Generator Program. Repair methods established in accordance with the Steam Generator Program are listed in Technical Specification 5.5.9 as in the 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.

Steam generator tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged-on-detection" and their integrity is assessed. This requirement is unchanged by the proposed technical specifications.

Note that SG plug designs do not require NRC review and therefore plugging is not considered a repair in the context of this requirement.

The proposed approach is not a change to the technical specifications.

Section 13: Reporting Requirements

The current technical specifications require the following reports:

- A report listing the number of tubes plugged or repaired in each SG submitted within 15 days of the end of the inspection.
- A SG inspection results report submitted within 12 months after the inspection.
- Reports required pursuant to 10 CFR 50.73.

The proposed change to Technical Specification 5.6.9 replaces the 15 day and the SG inspection reports with one report required within 180 days. The proposed report also contains more information than the current SG inspection report. This provision expands the report to provide more substantive information and will be sent earlier (180 days versus 12 months). This allows the NRC to focus its attention on the more significant conditions.

The guidance in NUREG-1022, Rev. 2, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," identifies serious SG tube degradation as an example of an event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. Steam generator tube degradation is considered serious if the tubing fails to meet the structural integrity or accident induced leakage performance criteria. Serious SG tube degradation would be reportable in accordance with 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) requiring NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence.

The proposed reporting requirements are an improvement as compared to those required by the current technical specifications. The proposed reporting requirements are more useful in identifying the degradation mechanisms and determining their effects. In the unlikely event that a performance criterion is not met, NEI 97-06 (Ref. 1) directs the licensee to submit additional information on the root cause of the condition and the basis for the next operating cycle.

The changes to the reporting requirements are performance based. The new requirements remove the burden of unnecessary reports from both the NRC and the licensee, while ensuring that critical information related to problems and significant tube degradation is reported more completely and, when required, more expeditiously than under the current technical specifications.

Section 14: SG Terminology

The proposed Steam Generator Tube Integrity specification Bases explain a number of terms that are important to the function of a Steam Generator Program. The Technical Specification Bases are controlled by the Technical Specification Bases Control Program, which appears in the Administrative Technical Specifications.

The terms are described below.

1. Accident induced leakage rate means the primary to secondary LEAKAGE rate occurring during postulated accidents other than a steam generator tube rupture. This includes the primary to secondary LEAKAGE rate existing immediately prior to the accident plus additional primary to secondary LEAKAGE induced during the accident.

   Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary to secondary leak rate during postulated design basis accidents must not cause radiological dose consequences in excess of the 10 CFR Part 100 guidelines for offsite doses, or the GDC-19 requirements for control room personnel, or other NRC approved licensing basis.

2. The LCO section of Steam Generator Tube Integrity Bases define the term "burst" 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."

   Since a burst definition is required for condition monitoring, a definition that can be analytically defined and is capable of being assessed via in situ and laboratory testing is necessary. Furthermore, the definition must be consistent with ASME Code requirements, and apply to most forms of tube degradation.

   The definition developed for tube burst is consistent with the testimony of James Knight (Ref. 9), and the historical guidance of draft Regulatory Guide 1.121 (Ref. 10). The definition of burst per these documents is in relation to gross failure of the pressure boundary; e.g., "the degree of loading required to burst or collapse a tube wall is consistent with the design margins in Section III of the ASME B&PV Code (Ref. 11)." Burst, or gross failure, according to the Code would be interpreted as a catastrophic failure of the pressure boundary.

   The above definition of burst was chosen for a number of reasons:

   - The burst definition supports field application of the condition monitoring process. For example, verification of structural integrity during condition monitoring may be accomplished via in situ testing. Since these tests do not have the capability to provide an unlimited water supply, or the capability to maintain pressure under certain leakage scenarios, opening area may be more a function of fluid reservoir rather than tube strength. Additionally, in situ designs with bladders may not be reinforced. In certain cases, the bladder may rupture when tearing or extension of the defect has not occurred. This condition may simply mean the opening of the flanks of the defect was sufficient to permit extrusion of the bladder, and that the actual, or true, burst pressure was not achieved during the test. The burst definition addresses this issue.

   - The definition does not characterize local instability or "ligament pop-through", as a burst. The onset of ligament tearing need not coincide with the onset of a full burst. For example, an axial crack about 0.5" long with a uniform depth at 98% of the tube wall would be expected to fail the remaining ligament, (i.e., extend the crack tip in the radial direction) due to deformation during pressurization at a pressure below that required to cause extension at the tips in the axial direction. Thus, this would represent a leakage situation as opposed to a burst situation and a factor of safety of three against crack extension in the axial direction may still be demonstrated. Similar conditions have been observed for localized deep wear indications.

3. The LCO section of Steam Generator Tube Integrity Bases define a SG tube as, "the entire length of the tube, including the tube wall and any repairs 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."

   This definition ensures that all portions of SG tubes that are part of the RCPB, with the exception of the tube-to-tubesheet weld, are subject to Steam Generator Program requirements. The definition is also intended to exclude tube ends that can not be NDE inspected by eddy current. If there are concerns in the area of the tube end, they will be addressed by NDE techniques if possible or by using other methods if necessary.

For the purposes of SG tube integrity inspection, any weld metal in the area of the tube end is not considered part of the tube. This is necessary since the acceptance requirements for tubing and weld metals are different.

4. The LCO section of Steam Generator Tube Integrity Bases define the term "collapse" 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."

   In dealing with pure pressure loadings, burst is the only failure mechanism of interest. If bending loads are introduced in combination with pressure loading, the definition of failure must be broadened to encompass both burst and bending collapse. Which failure mode applies depends on the relative magnitude of the pressure and bending loads and also on the nature of any flaws that may be present in the tube. Guidance on assessing applicable failure modes is provided in the EPRI steam generator guidelines.

5. The LCO section of Steam Generator Tube Integrity Bases define the term "significant" as used in the structural integrity performance criterion 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."

6. The LCO section of Steam Generator Tube Integrity Bases describes how to determine whether thermal loads are primary or secondary loads. 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.

Conclusion

The proposed changes will provide greater assurance of SG tube integrity than that offered by the current technical specifications. The proposed requirements are performance based and provide the flexibility to adopt new technology as it matures. These changes are consistent with the guidance in NEI 97-06, "Steam Generator Program Guidelines," (Ref. 1).

Adopting the proposed changes will provide added assurance that SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The proposed change revises the improved Standard Technical Specification (iSTS) definition of LEAKAGE, Section 3.4.13, "RCS Operational LEAKAGE," Section 5.5.9, "Steam Generator Tube Surveillance Program, and Section 5.6.9, "Steam Generator Tube Inspection Report," and adds a new specification for Steam Generator Tube Integrity. The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines," (Reference 1). The TSTF has evaluated whether or not a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No

The proposed change requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

The structural integrity performance criterion is:

> 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.

The accident induced leakage performance criterion is:

> The primary to secondary accident induced leakage rate for any design basis accidents, 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.

The operational LEAKAGE performance criterion is:

> The RCS operational primary to secondary LEAKAGE through any one SG shall be limited to 150 gallons per day.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change to the ISTS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the Steam Generator Program required by the proposed change to the ISTS. The program, defined by NEI 97-06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I-131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the plant technical specifications for operational leakage and for DOSE EQUIVALENT I-131 in primary coolant to ensure the plant is operated within its analyzed condition. The typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than 500 gallons per day or 720 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I-131 are at the technical specification values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current technical specifications and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current technical specifications.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed performance based requirements are an improvement over the requirements imposed by the current technical specifications.

Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment, repair, and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current technical specifications.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the ISTS.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory requirements applicable to SG tube integrity are the following:

10 CFR 50.55a, Codes and Standards - Section (b), ASME Code - c) *Reactor coolant pressure boundary.* (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

The proposed change and the Steam Generator Program requirements which underlie it are in full compliance with the ASME Code. The proposed technical specifications are more effective at ensuring tube integrity and, therefore, compliance with the ASME Code, than the current technical specifications as described in Section 4.0 (Technical Analysis).

10 CFR 50.65 Maintenance Rule – Each holder of a license to operate a nuclear power plant under §§50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in §50.82(a)(1), this section only shall apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that such structures, systems, and components are capable of fulfilling their intended functions.

Under the Maintenance Rule, licensees classify SGs as risk significant components because they are relied on to remain functional during and after design basis events. The performance criteria included in the proposed technical specifications are used to demonstrate that the condition of the SG "is being effectively controlled through the performance of appropriate preventive maintenance" (Maintenance Rule §(a)(2)). If the performance criteria are not met, a root cause determination of appropriate depth is done and the results evaluated to determine if goals should be established per §(a)(1) of the Maintenance Rule.

NEI 97-06, Steam Generator Program Guidelines, and its referenced EPRI guidelines define a SG program that provides the appropriate preventive maintenance that meets the intent of the Maintenance Rule. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Reference 13) offers guidance for implementing the Maintenance Rule should a licensee elect to incorporate additional monitoring goals beyond the scope of those documented in NEI 97-06.

10 CFR 50, Appendix A, GDC 14 – *Reactor Coolant Pressure Boundary.* The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

There are no changes to the SG design that impact this general design criteria. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

10 CFR 50, Appendix A, GDC 30 - *Quality of reactor coolant pressure boundary.* Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

There are no changes to the SG design that impact this general design criteria. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

10 CFR 50, Appendix A, GDC 32 – *Inspection of reactor coolant pressure boundary.* Components which are part of the reactor coolant pressure boundary shall be designed to (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

There are no changes to the SG design that impact this general design criteria. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

General Design Criteria (GDC) 14, 30, and 32 of 10 CFR Part 50, Appendix A, define requirements for the reactor coolant pressure boundary with respect to structural and leakage integrity. Steam generator tubing and tube repairs constitute a major fraction of the reactor coolant pressure boundary surface area. Steam generator tubing and associated repair techniques and components, such as plugs and sleeves, must be capable of maintaining reactor coolant inventory and pressure. The Steam Generator Program required by the proposed technical specification establishes performance criteria, repair criteria, repair methods, inspection intervals and the methods necessary to meet them. These requirements provide reasonable assurance that tube integrity will be met in the interval between SG inspections.

The proposed change provides requirements that are more effective in detecting SG degradation and prescribing corrective actions. The proposed change results in added assurance of the function and integrity of SG tubes. Therefore, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

2) Such activities will be conducted in compliance with the Commission's regulations; and

3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed change would change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

7.0 REFERENCES

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

2. EPRI, "Steam Generator Examination Guideline."

3. EPRI, "Steam Generator Integrity Assessment Guideline."

4. EPRI, "Steam Generator In-situ Pressure Test Guideline."

5. EPRI, "PWR Primary-to-Secondary Leak Guideline."

6. EPRI, "Primary Water Chemistry Guideline."

7. EPRI, "Secondary Water Chemistry Guideline."

8. EPRI Report R-5515-00-2, "Experience of US and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves," June 5, 2002.

9. Testimony of James Knight Before the Atomic Safety and Licensing Board, Docket Nos. 50-282 and 50-306, January 1975.

10. Draft Regulatory Guide 1.121, "Bases for Plugging Degraded Steam Generator Tubes," August 1976.

11. ASME B&PV Code, Section III, Rules for Construction of Nuclear Facility Components.

12. Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995.

13. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3.

14. S. C. Collins memo to W. D. Travers, "Steam Generator Action Plan Revision to Address Differing Professional Opinion on Steam Generator Tube Integrity," May 11, 2001.

INSERT 3.4.13 A

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

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, "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).

<u>INSERT B 3.4.13 E</u>

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

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

INSERT 5.5.9

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].

    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.]

INSERT 5.6.9

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

e. Number of tubes plugged [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,

[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.]

1.1 Definitions

LEAKAGE (continued)

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

primary to secondary

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

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

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$ — $F_Q(Z)$ shall be the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions.

NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR $F^N_{\Delta H}$ — $F^N_{\Delta H}$ shall be the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power.

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

3.4 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~~

150

d. ~~e.~~ ~~[[720]]~~ 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 S--- 1. Not required to be performed until 12 hours after establishment of 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.~~ 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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged [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. OR SG tube integrity not maintained.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged [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 Inservice Testing Program (continued)

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

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

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

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

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

Insert 5.5.9

~~-----REVIEWER'S NOTE-----~~
~~The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~
~~-----~~

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

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

5.6 Reporting Requirements

5.6.8 [Tendon Surveillance Report

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

5.6.9 Steam Generator Tube Inspection Report

Insert 5.6.9

~~-----REVIEWER'S NOTE-----~~

1. ~~Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~

2. ~~These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

BASES

LCO

The purpose of this LCO is to require two loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the required way to transport heat, although natural circulation flow provides adequate removal. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The Note permits a limited period of operation without RCPs. All RCPs may be removed from operation for ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal (DHR) System, and otherwise may be de-energized for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, boron reduction with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least [10]°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation (e.g., change operation from one DHR train to the other, to perform surveillance or startup testing, to perform the transition to and from DHR System cooling, or to avoid operation below the RCP minimum net positive suction head limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

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

Operation in other MODES is covered by:

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

BASES

LCO (continued)

The Note also permits the DHR pumps to be stopped for ≤ 1 hour per 8 hour period. When the DHR pumps are stopped, no alternate heat removal path exists, unless the RCS and SGs have been placed in service in forced or natural circulation. The response of the RCS without the DHR System depends on the core decay heat load and the length of time that the DHR pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by DHR, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) or low temperature overpressure protection (LTOP) limits) must be observed and forced DHR flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both DHR trains are to be limited to situations where:

a. Pressure and pressure and temperature increases can be maintained well within the allowable pressure (P/T and LTOP) and 10°F subcooling limits or

b. An alternate heat removal path through the SG is in operation.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~.

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

APPLICABILITY

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

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,	"Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" (MODE 6), and
LCO 3.9.5,	"Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" (MODE 6).

BASES

LCO (continued)

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

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting DHR loops to not be in operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the DHR loops.

An OPERABLE DHR loop is composed of an OPERABLE DHR pump and an OPERABLE DHR heat exchanger.

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

APPLICABILITY

In MODE 5 with loops filled, forced circulation is provided by this LCO to remove decay heat from the core and to provide proper boron mixing. One loop of DHR provides sufficient circulation for these purposes.

Operation in other MODES is covered by:

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

ACTIONS

A.1, A.2, B.1, and B.2

If one DHR loop is OPERABLE and any required SG has secondary side water level < [50]% or one required DHR loop inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second DHR loop to OPERABLE status or initiate action to restore the secondary side water level in the SGs, and action must be taken immediately. Either Required Action will restore redundant decay heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

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

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

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

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

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

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA). However, the ability to monitor leakage provides advance warning to permit plant shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Insert B3.4.13A

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. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The [1 gpm] primary to secondary LEAKAGE is relatively inconsequential.

safety analysis assumption

the entire

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes [1 gpm] primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100.

is through the affected

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

b. Unidentified LEAKAGE

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

BASES

LCO (continued)

c. Identified LEAKAGE

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

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

~~Total primary to secondary LEAKAGE amounting to 1 gpm 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.~~

Insert B 3.4.13B

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

~~The [720] gallon per day limit on one SG allocates the total 1 gpm allowed primary to secondary LEAKAGE equally between the two generators.]~~

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 leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

BASES

ACTIONS

A.1

or

If unidentified LEAKAGE, identified LEAKAGE, ~~or primary to secondary LEAKAGE~~ are in excess of the LCO limits, the LEAKAGE 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

or primary to secondary LEAKAGE is not within limits

or

If any pressure boundary LEAKAGE exists or if unidentified, identified, ~~or primary to secondary~~ LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging 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 within the LCO limits ensures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. 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 performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, [and RCP seal injection and return flows]). ~~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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

Insert B 3.4.13C

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

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

SR 3.4.13.2

Insert B 3.4.13D (BWOG)

~~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. FSAR, Chapter [15].

Insert B 3.4.13E

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

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

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

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

Specification 5.5.9, "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, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by 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.

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 5.5.9, "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.

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. <u>A.1 and A.2</u> 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 3.4.17.2. 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 3.4.17.1

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

<u>SR 3.4.17.2</u>

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 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."

1.1 Definitions

Ē - AVERAGE DISINTEGRATION ENERGY	Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

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)

1.1 Definitions

LEAKAGE (continued)

b. Unidentified LEAKAGE

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

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

OPERABLE – OPERABILITY

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

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter [14, Initial Test Program] of the FSAR,

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.~~ 150 ~~[500]~~ gallons per day primary to secondary LEAKAGE through any one steam generator (SG) ~~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 AND B.2	Be in MODE 3. 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	S -------------------NOTE------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. --- Verify RCS operational leakage is within limits by performance of RCS water inventory balance. LEAKAGE	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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.20 Steam Generator (SG) Tube Integrity

LCO 3.4.20 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
OR	AND	
SG tube integrity not maintained.	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.20.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.20.2	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 Inservice Testing Program (continued)

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

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

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

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

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

Insert 5.5.9

~~REVIEWER'S NOTE~~
~~The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

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

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

5.6 Reporting Requirements

5.6.7 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.

5.6.8 [Tendon Surveillance Report

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

5.6.9 ~~[~~ Steam Generator Tube Inspection Report ~~]~~

Insert 5.6.9

~~REVIEWER'S NOTE~~

~~1. Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~

~~2. These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

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, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RTP. This is the design overpower condition for [four] RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 107% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

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

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 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.

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.

BASES

LCO (continued)

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

a. No operations are permitted that would dilute the RCS boron concentration 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 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 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,"
LCO 3.9.5,	"Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6), and
LCO 3.9.6,	"Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

BASES

LCO (continued)

stopped for a short period of time. The Note permits the stopping of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

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

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentrations less than required to meet SDM of LCO 3.1.1, therefore 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 and

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

Note 2 requires that the secondary side water temperature of each SG be ≤ [50]°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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.

BASES

LCO (continued)

Note 3 requires that the secondary side water temperature of each SG be ≤ [50]°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ [275°F] [Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR]. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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

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~~.

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 ≥ [17]%.

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2;"
LCO 3.4.5,	"RCS Loops - MODE 3;"
LCO 3.4.6,	"RCS Loops - MODE 4;"
LCO 3.4.8,	"RCS Loops - MODE 5, Loops Not Filled;"
LCO 3.9.5,	"Residual Heat Removal (RHR) and Coolant Circulation High Water Level" (MODE 6)," and
LCO 3.9.6,	"Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6)."

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE 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. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The [1 gpm] primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The SLB is more limiting for the entire site radiation releases. The safety analysis for the SLB accident assumes [1 gpm] primary to secondary LEAKAGE [in one] 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 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

b. Unidentified LEAKAGE

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

BASES

LCO (continued)

c. Identified LEAKAGE

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

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

~~Total primary to secondary LEAKAGE amounting to 1 gpm 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.~~

Insert B 3.4.13B

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

~~The [500] gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.~~

APPLICABILITY

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

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

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

BASES

ACTIONS

A.1

or

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

B.1 and B.2

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 (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, [and RCP seal injection and return flows]). ~~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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

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

Insert B3.4.13C

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 (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. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [15].

Insert B 3.4.13 E

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.20 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

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

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

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

Specification 5.5.9, "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, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by 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. 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 5.5.9, "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.

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.

<u>A.1 and A.2</u>

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

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

The Steam Generator Program defines the Frequency of SR 3.4.20.1. 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 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

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 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."

1.1 Definitions

LEAKAGE (continued)

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System,

b. Unidentified LEAKAGE (primary to secondary LEAKAGE)

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

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.

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.

OPERABLE – OPERABILITY

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

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.

These tests are:

a. Described in Chapter [14, Initial Test Program] of the FSAR,

b. Authorized under the provisions of 10 CFR 50.59, or

c. Otherwise approved by the Nuclear Regulatory Commission.

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.~~ 150 ~~[720]~~ gallons per day primary to secondary LEAKAGE through any one ~~SG~~ 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	---NOTE--- (S) 1. Not required to be performed until 12 hours after establishment 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 SG 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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 SG tube integrity shall be maintained.

<u>AND</u>

All SG tubes satisfying the tube repair criteria shall be plugged [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. <u>AND</u> A.2 Plug [or repair] the affected tube(s) in accordance with the Steam Generator Program.	7 days 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. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.18.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged [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 Inservice Testing Program (continued)

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

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

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

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

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

Insert 5.5.9

~~REVIEWER'S NOTE~~

~~The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

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

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

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

5.6.9 Steam Generator Tube Inspector Report

Insert 5.6.9

~~REVIEWER'S NOTES~~

1. ~~Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~

2. ~~These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming four RCPs are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are of most importance to 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 had been performed for the [four] pump combination. For [four] pump operation, the steady state DNB analysis, which generates the pressure and temperature and Safety Limit (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of 107% RTP. This is the design overpower condition for four pump operation. The 107% value is the accident analysis setpoint of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

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

LCO

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

Each OPERABLE loop consists of two RCPs providing forced flow for heat transport to an SG that is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~. SG, and hence RCS loop, OPERABILITY with regard to SG water level is ensured by the Reactor Protection System (RPS) in MODES 1 and 2. A reactor trip places the plant in MODE 3 if any SG level is ≤ [25]% as sensed by the RPS. The minimum water level to declare the SG OPERABLE is [25]%.

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)

The Note permits a limited period of operation without RCPs. All RCPs may be removed from operation for ≤ 1 hour per 8 hour period. This means that natural circulation has been established. When in natural circulation, a reduction in boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCPs or shutdown cooling (SDC) pump forced circulation (e.g., to change operation from one SDC train to the other, to perform surveillance or startup testing, to perform the transition to and from SDC System cooling, or to avoid operation below the RCP minimum net positive suction head limit). The time period is acceptable because natural circulation is adequate for heat removal, or the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~. A RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

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

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2,"
LCO 3.4.6,	"RCS Loops - MODE 4,"
LCO 3.4.7,	"RCS Loops - MODE 5, Loops Filled,"
LCO 3.4.8,	"RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4,	"Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" (MODE 6), and
LCO 3.9.5,	"Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" (MODE 6).

BASES

LCO (continued)

pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by forced flow, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) limits or low temperature overpressure protection (LTOP) limits) must be observed and forced SDC flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. The circumstances for stopping both RCPs or SDC pumps are to be limited to situations where:

a. Pressure and temperature increases can be maintained well within the allowable pressure (P/T limits and LTOP) and 10°F subcooling limits or

b. An alternate heat removal path through the SGs is in operation.

Note 2 requires that either of the following two conditions be satisfied before an RCP may be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR:

a. Pressurizer water level is < [60]% or

b. Secondary side water temperature in each SG is < [100]°F above each of the RCS cold leg temperatures.

Satisfying either of the above conditions will preclude a large pressure surge in the RCS when the RCP is started.

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

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

BASES

LCO (continued)

Note 3 requires that either of the following two conditions be satisfied before an RCP may be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR:

a. Pressurizer water level must be < [60]% or

b. Secondary side water temperature in each SG must be < [100]°F above each of the RCS cold leg temperatures.

Satisfying either of the above conditions will preclude a low temperature overpressure event due to a thermal transient when the RCP is started.

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

An OPERABLE SDC train is composed of an OPERABLE SDC pump and an OPERABLE SDC heat exchanger.

SDC pumps are OPERABLE if they are capable of being powered and are able to provide 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 SG Tube Surveillance Program~~.

APPLICABILITY

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

Operation in other MODES is covered by:

LCO 3.4.4,	"RCS Loops - MODES 1 and 2,"
LCO 3.4.5,	"RCS Loops - MODE 3,"
LCO 3.4.6,	"RCS Loops – MODE 4,"
LCO 3.4.8,	"RCS Loops - MODE 5, Loops Not Filled,"
LCO 3.9.4,	"Shutdown Cooling (SDC) and Coolant Circulation - High Water Level" (MODE 6), and
LCO 3.9.5,	"Shutdown Cooling (SDC) and Coolant Circulation - Low Water Level" (MODE 6).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes ~~a 1 gpm primary to secondary LEAKAGE 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. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The [1 gpm] primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The SLB is more limiting for the entire site radiation releases. The safety analysis for the SLB accident assumes [gpm] primary to secondary LEAKAGE ~~in one~~ 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 or the staff approved licensing basis (i.e., a small fraction of these limits).

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

b. Unidentified LEAKAGE

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

BASES

LCO (continued)

c. Identified LEAKAGE

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

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 leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

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

~~Total primary to secondary LEAKAGE amounting to 1 gpm 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 analysis. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

Insert B 3.4.13B

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

~~The [720] gallon per day limit on primary to secondary LEAKAGE through any one SG allocates the total 1 gpm allowed primary to secondary LEAKAGE equally between the two generators.~~

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.

BASES

ACTIONS

A.1

or

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

B.1 and B.2

Or primary to secondary LEAKAGE is not within limit

or

If any pressure boundary LEAKAGE exists or if unidentified/identified~~, or primary to secondary~~ LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the 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. 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 performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, [and RCP seal injection and return flows]). ~~Therefore, a~~ Note ~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

1 states

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. 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 (CEOG)

~~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. FSAR, Section [15].

Insert B3.4.13E

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

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

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

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

Specification 5.5.9, "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, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by 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. 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 5.5.9, "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.

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.

<u>A.1 and A.2</u>

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 3.4.18.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged [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 3.4.18.1

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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

The Steam Generator Program defines the Frequency of SR 3.4.18.1. 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 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

<u>SR 3.4.18.2</u>

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 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."