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NUCLEAR ENERGY INSTITUTE

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February 4, 2000

Mr. Samuel J. Collins
Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Industry Steam Generator Program Generic License Change Package

PROJECT NUMBER: 689

Dear Mr. Collins:

In 1997, the industry adopted a formal industry position to achieve uniform, safe and reliable steam generator performance at each pressurized water reactor site via implementation of NEI 97-06, *Steam Generator Program Guidelines*. That initiative focuses on consistent use of industry-developed guidelines related to managing steam generator programs.

During the last year, the industry and the NRC worked together to develop a regulatory framework for the steam generator program content in NEI 97-06. The result of these efforts is the enclosed Steam Generator Program Generic License Change Package. Enclosures 1 through 6 will be used as templates for plant-specific license amendment requests. NEI is seeking the NRC's endorsement of these documents.

The enclosed Generic License Change Package relocates much of the existing steam generator Standard Technical Specifications (STS) surveillance and limiting conditions for operation requirements to other licensee controlled documents. This approach is consistent with the NRC policy established when developing the improved STS and promulgated in the 1995 change to § 50.36, *Technical Specifications*. In the improved STS, many technical specification sections were modified to relocate program requirements not directly of controlling importance to the plant operating staff. The changes in the Steam Generator Program Generic License Change Package are structured similarly, while maintaining an appropriate level of NRC oversight and ensuring the ability of licensees to adopt improved technologies and guidelines.

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The industry and the NRC have cooperated closely in the development of the Steam Generator Program Generic License Change Package. The Package has been reviewed by the industry and remains substantially unchanged from the version developed with the NRC. As a result, no technical issues should remain to be resolved.

Once the NRC approves the enclosed Generic License Change Package, it is expected that PWR licensees will use the documents as templates in the preparation of plant-specific license amendment requests that meet the intent of the requirements in the Generic Package.

Implementation of the content of the Generic License Change Package represents an improvement in the safe operation of PWR power plants, and the NRC and industry are fundamentally in agreement with the content of this Package. For these reasons, there is both an incentive to obtain NRC approval as rapidly as possible, and no known obstacles to prevent achieving this goal. Therefore, the industry looks forward to a rapid response from the NRC in support of our initiative, and anticipates your approval of this Package by July 1, 2000.

A revision to NEI 97-06 will be necessary to reflect the resolution of the technical issues addressed during the last year. A draft of revision 1 is enclosed with this letter to provide background information so that the regulatory requirements can be understood within the context of the entire steam generator program. Because the content of revision 1 is dependent on the technical specification provisions in the Generic License Change Package, revision 1 will not be approved by the industry until after the proposed technical specifications are approved by the NRC. We are not seeking NRC approval of revision 1 to NEI 97-06.

The industry expects to participate in a continuing dialog with the staff as the NRC completes its review and approval of the enclosed documents. As a first step in this process, a senior management meeting in March would be useful to reassert our mutual commitment to this effort and to communicate our expectations with respect to its completion.

This submittal provides information that might be helpful to NRC staff when evaluating plant-specific license amendment requests. Such reviews are exempted under §170.21, Schedule of Facility Fees. Footnote 4 to the Special Projects provision of §170.21 states, "Fees will not be assessed for requests/reports submitted to the NRC...[a]s means of exchanging information between industry organizations and the NRC for the purpose of supporting generic regulatory improvements or efforts."

We look forward to our next meeting on this subject. Please call Jim Riley (202-739-8137) if you have any questions regarding these matters.

Sincerely,

David J. Modeen

JHR/edb
Enclosures:

1. Template for a Plant Specific License Amendment Cover Letter
2. Template for an Operational Leakage Technical Specification
3. Template for an Administrative Section Technical Specification
4. Template for a Licensee Controlled Document (Technical Requirements Manual)
5. Template for a License Amendment Safety Analysis
6. Template for a License Amendment Significant Hazards Consideration
7. Draft Revision 1 to NEI 97-06, Steam Generator Program Guidelines

c: Dr. Brian W. Sheron, U.S. Nuclear Regulatory Commission
Mr. Jack R. Strosnider, Jr., U.S. Nuclear Regulatory Commission
Mr. Edmund J. Sullivan, U.S. Nuclear Regulatory Commission
Mr. James W. Andersen, U.S. Nuclear Regulatory Commission
Mr. Stewart L. Magruder, Jr., U.S. Nuclear Regulatory Commission

Enclosure 1

Template for a Plant Specific License Amendment Cover Letter

[Month Day, 2000]

U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: [Plant, Docket Number]
License Amendment Request: Revision to Steam Generator Technical Specifications

REFERENCES:

Pursuant to 10 CFR 50.90, [Licensee] hereby requests an amendment to Operating License Number [DPR-XX] to incorporate the changes described below into the Technical Specifications for [Plant].

The proposed amendment removes the steam generator Technical Specification [3.4.6] and revises the Technical Specifications for RCS Operational Leakage [3.4.13], Steam Generator Program [5.5.9], and Steam Generator Tube Inspection Report [5.6.10].

The proposed amendment is necessary in order to implement the requirements of the Industry Initiative on NEI 97-06, Steam Generator Program Guidelines. The proposed changes reflect the results of a series of meetings between the NRC Staff and The Nuclear Energy Institute's Steam Generator Task Force.

This amendment request provides a programmatic framework for monitoring and maintaining the integrity of the steam generator tubes consistent with Appendices A and B to 10 CFR Part 50 and [Plant's] licensing basis. This framework includes performance criteria that, if satisfied, provide reasonable assurance that tube integrity is being maintained consistent with the licensing basis. In addition, this framework provides for monitoring and maintaining the tubes to provide reasonable assurance that the performance criteria are met at all times between scheduled inspections of the tubes.

DESCRIPTION OF PROPOSED CHANGE

Steam generator Technical Specification [3.4.6] is deleted by this request. The requirements of Technical Specification [3.4.6] are revised and relocated into a licensee controlled document and Technical Specifications [3.4.13, 5.5.9, and 5.6.10] as described below. The licensee controlled document, [*Technical Requirements Manual*], defines the approved steam generator performance criteria, repair criteria, and repair methods and establishes actions that would be necessary should the performance criteria not be met. The [*Technical Requirements Manual*] also contains definitions pertinent to steam generator program requirements. Changes to the [*Technical Requirements Manual*] will be governed by the requirements of 10 CFR 50.59

The changes to the Operational Leakage Technical Specification reduce the allowable leakage from any one steam generator to [*150 gallons per day*] and reference the plant's steam generator program described in Technical Specification [5.5.9] for the surveillance requirements necessary to verify tube integrity.

The changes to Administrative Technical Specification [5.5.9], Steam Generator Program, require the implementation of a steam generator program that contains the steam generator performance criteria. In addition, this section defines the approval process for revising the performance criteria, tube repair criteria and repair methods.

Finally, the change to Technical Specification [5.6.10] defines the requirement for, and contents of the steam generator tube inspection report. The existing requirement for a twelve month report is changed to a 120 day report, submitted only if the number of tubes exceeding the repair criteria during scheduled inservice inspections exceeds 1 percent of those inspected.

The content of the steam generator program as discussed in this submittal is critical to the satisfactory maintenance of steam generator tube integrity. [Plant's] steam generator program will meet the intent of the guidance provided in the Steam Generator Integrity Elements section of NEI 97-06, Steam Generator Program Guidelines, as it may be revised from time to time. The basis for any deviations from NEI 97-06 or its referenced EPRI guideline documents will be documented internally as part of the program implementation. This approach will be documented as a commitment in [Plant's] [Commitment Tracking System].

This proposed revision will enhance the safety function of the steam generators by increasing the probability that the integrity of the steam generator tubes will be maintained between scheduled inservice inspections.

REQUESTED CHANGES

Revise Technical Specifications [3.4.6, 3.4.13, 5.9.5, and 5.6.10] as shown in the attached marked-up Technical Specifications pages in Enclosure (3).

SCHEDULE

Approval of the proposed technical specification amendment is requested by [January 1, 2001] in order to allow implementation of the associated requirements for scheduled refueling outages after [X].

ASSESSMENT AND REVIEW

[Licensee] has evaluated the significant hazards considerations associated with the proposed license amendment, as required by 10 CFR 50.92, and has determined that there are none (see Enclosure (2) for a complete discussion). [Licensee] has also determined that operation with the proposed changes will not result in any significant increases in the amounts of any effluents that may be released offsite, and no significant increases in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change. The [Plant Operations and Safety Review Committee] has reviewed this proposed amendment and concur that operation with the proposed modification will not result in an undue risk to the health and safety of the public.

Should you have any questions regarding this matter, we will be pleased to discuss them with you.

Very Truly Yours,

Enclosures: ( ) Summary Description and Safety Analysis
( ) Determination of Significant Hazards
( ) Technical Specification Marked-up Pages

Enclosure 2

Template for an Operational Leakage Technical Specification

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

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

d ~~e.~~ [~~500~~ 150] 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.	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.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

---------------NOTE---------------
Not applicable to primary-to-secondary leakage

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-------------NOTE------------- Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. ------------------------------ Perform RCS water inventory balance.	-----NOTE------ Only required to be performed during steady state operation -------------- 72 hours
SR 3.4.13.2	Verify ~~steam generator tube integrity~~ primary to secondary leakage is in accordance with the Operational Leakage performance criteria described in the Steam Generator ~~Tube Surveillance~~ Program.	In accordance with the Steam Generator ~~Tube Surveillance~~ Program

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

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

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

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

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

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

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

(continued)

BASES (continued)

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 from following page

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~~ 150 gpd primary to secondary LEAKAGE is relatively inconsequential.

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 or the staff approved licensing basis (i.e., a small fraction of these limits).

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

(continued)

B LCO 3.4.13 RCS Operational LEAKAGE

APPLICABLE SAFETY ANALYSIS (insert):

The Technical Specification requirement to limit primary to secondary leakage through any one steam generator to less than 150 gallons per day is significantly less than the initial condition of the safety analysis. A limit of 150 gallons per day is based on operating experience as an indication of one or more propagating tube leak mechanisms. This leakage rate provides reasonable assurance against tube burst at normal and faulted conditions and provides reasonable assurance that flaws will not propagate to burst prior to detection by leakage monitoring methods and commencement of plant shutdown.

SR 3.4.13.1 (insert)

A note under the surveillance column states that this SR is not applicable to primary-to-secondary leakage because leakage limits as low as 150 gallons per day cannot be measured accurately by RCS inventory balance.

SR3.4.13.2

This SR requires the determination of SG OPERABILITY in an operational MODE. The requirement to demonstrate SG OPERABILITY includes verification that the Operational Leakage performance criterion in the steam generator program is satisfied. This surveillance requirement emphasizes the importance of SG tube integrity and provides reasonable assurance of tube integrity under operating conditions. The 150 gallons per day limit is measured at standard temperature and pressure.

BASES

LCO (continued)

b. Unidentified LEAKAGE

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

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of identified 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.~~

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

The ~~[500]~~ 150 gallons per day limit on one SG is based on the assumption that a single ~~crack~~ flaw 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~~ flaws are very small, and the above assumption is conservative.

BASES (continued)

APPLICABILITY

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

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

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

ACTIONS

A.1

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

B.1 and B.2

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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.~~

Insert from previous page

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

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

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the 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."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

SR 3.4.13.2

Insert from previous page

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].

Enclosure 3

Template for an Administrative Section Technical Specification for a Steam Generator Program

5.5.9 Steam Generator Program

5.5.9.1 General Requirements

A steam generator program shall be established and implemented to ensure that steam generator tube integrity is maintained. Steam generator tube integrity is maintained by meeting the performance criteria as defined in the steam generator program.

5.5.9.2 Condition Monitoring Assessment

Condition Monitoring Assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural and accident leakage integrity. The "as found" condition refers to the condition of the tubing during a steam generator 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 steam generator tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met. Requirements for condition monitoring are defined in the steam generator program.

5.5.9.3 Performance Criteria

The steam generator performance criteria are defined in the steam generator program. The licensee may revise its performance criteria (and their associated definitions as used in the steam generator program) after review and approval by the NRC. In addition, the licensee may revise its performance criteria (and their associated definitions as used in the steam generator program) to incorporate changes approved generically by the NRC subject to the limitations and conditions set forth in the staff's approving document. The licensee's demonstration of its satisfaction of the generic limitations and conditions must be documented in a safety evaluation prepared in accordance with 10 CFR 50.59.

5.5.9.4 Tube Repair Criteria and Repair Methods

Tube repair criteria and repair methods shall be described in and implemented by the steam generator program. The licensee may revise its repair criteria and repair methods after review and approval by the NRC. In addition, the licensee may use repair criteria and repair methods approved generically by the NRC subject to the limitations and conditions set forth in the staff's approving document. The licensee's demonstration of its satisfaction of the generic limitations and conditions must be documented in a safety evaluation prepared in accordance with 10 CFR 50.59. Note that tube plugging is not a repair and does not need to be reviewed or approved by the NRC.

5.6.10 Steam Generator Tube Inspection Report

If the results of the steam generator inspection indicate greater than 1% of the inspected tubes in any steam generator exceed the repair criteria in accordance with the requirements of the steam generator program, the licensee shall submit a special report within 120 days after the initial entry into Mode 4 following completion of the inspection. The report shall summarize:

a) The scope of inspections performed on each steam generator inspected in the affected unit during the current outage,
b) Active degradation mechanisms found,
c) NDE techniques utilized for each degradation mechanism,
d) Location, orientation(if linear) and measured sizes of service induced indications,
e) Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,
f) Repair method utilized and the number of tubes repaired by each repair method,
g) Total number and percentage of tubes plugged and/or repaired to date,
h) The effective plugging percentage for all plugging and tube repairs in each steam generator; and
i) The results of condition monitoring including the results of tube pulls and in-situ testing.

Enclosure 4

Template for a Licensee Controlled Document [Technical Requirements Manual]

The content of this document must be adopted when implementing the proposed license changes. The format used herein is not required.

TRM x.y Steam Generators

Each steam generator shall meet primary to secondary pressure boundary integrity performance criteria as given below during Modes 1, 2, 3, and 4.

Performance Criteria

(i) Structural criterion:
Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a margin of 3.0 against **burst** under **normal steady state full power operation** and a margin of 1.4 against **burst** under the **limiting design basis accident**, concurrent with a safe shutdown earthquake.

(ii) **Accident induced leakage** criterion:
The primary to secondary **accident induced leakage** rate for the **limiting design basis accident**, 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 steam generator, except for specific types of degradation at specific locations where the tubes are confined, as approved by the NRC and enumerated in conjunction with the list of approved repair criteria in the TRM].

(iii) Operational leakage criterion:
The RCS operational primary to secondary leakage through any one steam generator shall be limited to 150 gallons per day at standard temperature and pressure.

Requirements related to the Operational Leakage criterion are delineated in the RCS Operational LEAKAGE Technical Specification.

[CONTINGENCY MEASURES:]

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Does not meet **repair criteria** as discovered in MODE 5 or 6	Plug or repair the tubes exceeding the repair criteria in accordance with **repair methods**	Prior to entering MODE 4

The content of this document must be adopted when implementing the proposed license changes. The format used herein is not required.

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Does not meet performance criteria (i) or (ii) as discovered in MODE 5 or 6	Notify the NRC of the failure to meet the performance criteria. AND Investigate to determine causal factors and implement corrective action plan AND Submit a report to the NRC including the performance criteria exceeded, a causal factor determination, and the corrective action plan. AND Submit a special report to the NRC describing the basis for the planned operating period.	In accordance with §50.72 and 50.73 Prior to entering MODE 4 In accordance with §50.73 120 days after entering MODE 4
C. Failure to implement required plugging or repair discovered while in MODE 1, 2, 3, or 4	Notify the NRC of the failure to implement required plugging or repair. AND Determine steam generators remain acceptable for continued operation based on meeting the performance criteria AND Submit a report to the NRC providing a causal factor determination, corrective actions taken, and the basis for the planned operating period.	In accordance with §50.72 and 50.73 In accordance with [the licensee's corrective action program]. In accordance with §50.73

[VERIFICATION REQUIREMENTS[

VERIFICATION	FREQUENCY
SR x.y Verify steam generator tube integrity is in accordance with the performance criteria described in the Steam Generator Program.	In accordance with the Steam Generator Program.

The content of this document must be adopted when implementing the proposed license changes. The format used herein is not required.

TRM x.z Definitions

The following definitions are applicable to Technical Requirement x.y only.

Accident induced leakage rate means the primary-to-secondary leakage 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.

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.

Limiting design basis accident is defined as the accident that results in either the largest differential pressure for structural considerations or the largest dose for accident leakage considerations.

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 necessary.

Repair Criteria are those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging. The repair criteria approved for use are:

- 40% nominal tube wall thickness (includes plug or repair on detection)
- [list Repair Criteria that are currently approved for licensee use]

Repair Methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a steam generator tube is not a repair. The repair methods approved for use are:

- [Laser welded sleeves (Reference:.....)]
- [TIG welded sleeves (Reference ....)]
- [list Repair Methods that are currently approved for licensee use]

Steam generator tubing refers to the entire length of the tube 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.

Enclosure 5

Template for a License Amendment Safety Analysis

A. Introduction

In December of 1998, the NRC Staff acknowledged that the program described by NEI 97-06 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). For the next twelve months the industry and the NRC participated in a series of meetings to resolve the differences and develop the regulatory framework necessary to implement a comprehensive steam generator program. This license amendment request is the culmination of that effort.

B. Background

The steam generator (SG) tubes in pressurized water reactors have a number of important safety functions. SG 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 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. Finally, the SG tubes may be relied upon to maintain their integrity under conditions resulting from core damage severe accidents consistent with the containment objectives of preventing uncontrolled fission product release.

Tube integrity means that the tubes are capable of performing their intended safety functions consistent with their 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 that occur throughout the industry. 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.

NDE techniques are used to measure the extent of tube degradation. When the degradation of the tube wall reaches a prescribed repair criteria, the tube is considered defective and corrective action is taken.

The criteria governing structural integrity of SG tubes were developed in the 1970s from assumptions relative to 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 PWR Technical Specifications (Tech Specs) 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 overly 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.

In addition to their reliance on the through wall repair criteria, the current standard Technical Specifications do not reflect the current inspection techniques, do not meet current industry practice for inspection scope, and do not represent a performance based approach to steam generator program requirements.

For the last several years, the industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program, has developed a generic approach to improving steam generator 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 the most appropriate NDE parameters.

As a result, the assurance of steam generator 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 cover topics such as:

- Steam Generator Examination
- Steam Generator Integrity Assessment
- In-situ Pressure Testing
- Steam Generator Tube Plug Assessment
- Sleeving Assessment
- Primary to Secondary Leakage
- Primary Water Chemistry, and
- Secondary Water Chemistry

These EPRI Guidelines, along with the upper tier document (NEI 97-06, "Steam Generator Program Guidelines") that ties the entire steam generator program together, define a comprehensive, performance based approach to managing steam generator performance.

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.

This license amendment request provides the integrated approach for addressing SG tube integrity.

C. Description of Amendment Request

The proposed amendment revises the Standard Technical Specifications for RCS Operational Leakage [3.4.13], Steam Generator Program [5.5.9] and Steam Generator Tube Inspection Report [5.6.10]. The amendment also includes a [*Technical Requirements Manual*] section on Steam Generators. These changes are intended to replace the existing Steam Generator Technical Specification in its entirety. Marked up Technical Specification pages appear in Attachment [*X*].

The changes to the Operational Leakage Technical Specification reduce the allowable leakage from any one steam generator to [*150 gallons per day*] and reference the plant's steam generator program required by Technical Specification [5.5.9.1] for the surveillance requirements necessary to verify tube integrity.

The changes to Administrative Technical Specification [5.5.9] require the implementation of a steam generator program and the existence of a [*Technical Requirements Manual*] that contains the steam generator performance criteria. In addition, this section defines the approval process for revising steam generator performance criteria, tube repair criteria and repair methods.

The change to Technical Specification [5.6.10] defines the requirement for, and contents of the steam generator tube inspection report.

Finally, this amendment request includes a [*Technical Requirements Manual*] that establishes steam generator performance criteria and defines actions that would be necessary should the criteria not be met. The [*Technical Requirements Manual*] also contains definitions pertinent to steam generator program requirements. Changes to the [*Technical Requirements Manual*] will be governed by the requirements of 10CFR50.59

The combination of these changes will implement the requirements of NEI 97-06.

D. Description of Proposed Changes to Steam Generator Requirements

The following table summarizes steam generator operation under the current licensing basis (assumed to be Westinghouse Standard Technical Specifications) and under the proposed license amendment.

Condition or Requirement	Current Licensing Basis	Proposed License Amendment	Note
Operational primary to secondary leakage	[< 1 gpm total through all SGs and <500 gpd through any one SG]	≤150 gpd] through any one SG	1
RCS leakage 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]	Unchanged	2
Frequency of verification of tube integrity	6 to 40 months depending on SG category defined by previous inspection results	Dependent on the previous inspection results and the anticipated defect growth rate	3
Tube sample selection	Based on SG Category, industry experience, random selection, existing indications, and results of the initial sample set - 3% of all tubes as a minimum	Dependent on a pre-outage evaluation of actual degradation locations and mechanisms	4
Inspection techniques	Not specified	Performance based - dependent on a pre-outage evaluation of degradation types and NDE technique capabilities	5

Condition or Requirement	Current Licensing Basis	Proposed License Amendment	Note
Performance criteria	Operational leakage [<1 gpm total or < 500 gpd for any one SG]. No criteria specified for structural integrity or accident induced leakage.	Structural integrity, accident induced leakage, and operational leakage criteria dependent on design basis limits. Condition monitoring assessment performed to verify compliance.	6
Repair criteria	Plug or repair tubes with imperfections extending [>40%] through wall or alternate criteria approved by NRC	Requirements unchanged, but NRC approval is only required for first time use.	7
Failure to Meet Performance or Repair Criteria	Performance Criteria not defined. Plug or repair tubes exceeding repair criteria.	Requirements established for failure to meet performance criteria. Plug or repair tubes exceeding repair criteria.	8
Repair methods	Methods (except plugging) require previous approval by the NRC	Unchanged except that NRC approval is required for first time use.	9
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.]	NRC reports required upon failure to meet a perform-ance criterion or dis-covery of failure to implement required plugging or repair, and/or 120 days after the initial entry into Mode 4 if >1% of the inspected tubes in any affected SG exceed repair.criteria	10

Condition or Requirement	Current Licensing Basis	Proposed License Amendment	Note
Definitions	Normal Tech Spec definitions did not address SG Program issues.	Applicable SG Program definitions added to the {Technical Requirements Manual}.	11

Further explanation of the information presented in the table above is provided in the following notes referenced in the table.

1. Operational Leakage

The primary to secondary leakage limit has been reduced to [≤150 gpd]. This leakage rate limit provides added assurance against tube rupture at normal operating and faulted conditions. 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 10CFR100 and GDC 19 dose limits 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. Industry guidelines for primary to secondary leakage have been issued which, if followed, ensure leakage is effectively monitored and timely action is taken before a leaking tube exceeds the performance criteria. The industry guidelines include additional criteria for unit shutdown if a rapidly increasing leak rate is detected.

The Technical Specification requirement to limit primary to secondary leakage through any one steam generator to less than [or equal to 150 gallons per day] is significantly less than the initial condition of the safety analysis, [1 gpm] which defined the value in the existing Technical Specifications.

2. Operational Leakage Actions

These actions are unaffected by the proposed change.

3. Frequency of Verification of Tube Integrity

The existing Technical Specifications base inspection intervals on classifying the condition of the steam generator into one of three categories based on the overall results of the previous inspection. The surveillance frequency in the proposed version is more performance based.

It is dependent on the location and severity of specific active degradation mechanisms and their anticipated growth rate.

The proposed amendment is an improvement over the existing Technical Specification. In the proposed Technical Specifications, the time between steam generator inservice inspections is variable. It is adjusted to minimize the possibility that tube integrity might degrade during the operating cycle beyond the limits defined by the performance criteria.

4. Tube Sample Selection

The existing Technical Specifications base tube selection on steam generator conditions and industry and plant experience. NEI 97-06 requires the performance of a degradation assessment before every steam generator inspection and refers utilities to EPRI Steam Generator Examination Guidelines and EPRI Steam Generator Integrity Assessment Guidelines for the details on degradation assessment. Tube sample selection is variable. The proposed approach is performance based, dependent upon actual steam generator conditions. 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 method proposed is an improvement over the existing Technical Specifications.

5. Inspection Techniques

NEI 97-06 requires the performance of a degradation assessment before every steam generator inspection and refers utilities to EPRI Steam Generator Examination Guidelines and EPRI Steam Generator Integrity Assessment Guidelines for the details on its performance. The degradation assessment will identify current and potential new degradation modes and mechanisms and the NDE techniques that are most effective in detecting their existence. 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.

This change is an improvement over the existing Technical Specifications that contained no similar requirement.

6. Performance Criteria

Performance criteria used for steam generators are based on tube structural integrity, accident induced leakage, and operational leakage. The proposed structural integrity and accident induced leakage performance criteria are new requirements. These criteria are documented in the [Technical Requirements Manual]. The existing Technical Specifications contained only the operational leakage criteron. The steam generator performance criteria identify the standards against which performance is to be measured. Meeting the performance criteria provides reasonable assurance that the steam generator tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle.

The **structural integrity** performance criterion is:

> *Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation and a safety factor of 1.4 against burst under the most limiting design basis accident for structural integrity considerations, concurrent with a safe shutdown earthquake.*

The structural performance criterion is based on ensuring that there is reasonable assurance that a steam generator tube will not **burst** during normal or postulated accident conditions.

In addition to the safety factor of three (3) for normal steady state operation and 1.4 for accident pressures, the integrity evaluation shall verify that the primary pressure stresses not exceed the yield strength for the full range of normal operating conditions as described in the performance criteria. Additionally, all appropriate loads contributing to combined primary plus secondary stress shall be evaluated so as to ensure that these loads do not significantly reduce the burst pressure for the full range of normal operating conditions including postulated accidents. For example, axial loads due to tube-to-shell temperature differences in once-through steam generator designs during postulated MSLB, or axial loading associated with locked tube supports in recirculating steam generator designs should be addressed to ensure that the types of degradation evaluated are not adversely impacted by these conditions.

The **accident-induced leakage** performance criterion is:

> *The primary to secondary accident induced leakage rate for the limiting design basis accident, 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 steam generator, except for specific types of degradation at specific locations where the tubes are confined, as approved by the NRC and enumerated in conjunction with the list of approved repair criteria in the TRM].

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 exceed the offsite radiological dose consequences required by 10 CFR Part 100 guidelines, or the radiological consequences to control room personnel required by GDC-19.

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. [Plant specific assumptions for accident induced leakage are defined in each licensee's licensing basis.] Probabilistic safety analysis sensitivity studies have shown that severe accident risk is sensitive to certain design basis parameters such as 1 gpm accident induced leakage. Leakage rates greater than 1 gpm per steam generator could possibly cause failure in adjacent tubes under the conditions associated with severe accident scenarios. As a result, leakage greater than [Plant's] design basis or 1 gpm per steam generator is not allowed.

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 should commence if primary-to-secondary leakage exceeds 150 gallons per day (GPD) at standard temperature and pressure conditions from any one steam generator.

The proposed amendment provides performance-based regulatory oversight of the steam generator program. 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 structural and accident leakage criteria were developed deterministically and are consistent with the [Plant] licensing basis. The operational leakage criteria was based on providing added assurance against tube rupture at normal operating and faulted conditions, as discussed above.

A change in any of the criteria will require prior NRC approval. Under the proposed licensing approach, the NRC Staff would approve each performance criterion before it was used the first time on either a plant specific basis or generically. In performing a generic review of the performance criterion, the Staff would designate the requirements that were instrumental in their approval of the proposal. Other plants that meet these requirements could implement the generically approved performance criterion under 10CFR50.59. Following the determination that all requirements were met, the [Technical Requirements Manual] could be revised by the utility to reflect approval of the criterion.

The proposed performance criteria are an improvement over the existing Technical Specifications that include only the Operational Leakage Criterion. Meeting the performance criteria provides reasonable assurance that the steam generator tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle.

7. <u>Repair Criteria</u>

The proposed Technical Specifications require that tubes that exceed approved tube repair criteria be repaired in accordance with approved methods. For plants experiencing a damage form or mechanism for which no depth sizing capability exists, tubes identified with such damage are "repaired/plugged-on-detection" and integrity should be assessed. This requirement is unchanged from the existing Technical Specifications. However, the means of obtaining NRC approval of new repair criteria is changed by this amendment.

Under the proposed Technical Specifications and [Technical Requirements Manual], repair criteria are established in [Plants] steam generator program and, once approved, are listed in the [Technical Requirements Manual]. Tubes with defects exceeding the repair criteria are plugged or repaired with one of the approved methods, also listed in the [Technical Requirements Manual].

The NRC Staff currently approves repair criteria on a plant-by-plant basis. The Staff reviews each plant specific license amendment request and approves the proposal on a plant specific basis, including a plant specific technical specification change which, as a minimum, lists each repair criterion and may include specific technique requirements.

Under the proposed licensing approach, the NRC Staff would continue to have the opportunity to approve each repair criterion before it is used the first time. However, in reviewing the repair criterion, the Staff would be able to review the changes for generic application, designating the requirements that were instrumental in their approval of the proposal. Other plants that meet these requirements could implement the generically approved repair criterion under 10CFR50.59. Following the determination that all requirements were met, the [Technical Requirements Manual] could be revised by the utility to reflect approval of the criterion or method.

In summary, the NRC Staff would continue to have the opportunity to approve all repair criteria that are implemented by domestic PWRs. However, the proposed approach would eliminate the wasteful expenditure of resources in the review of plant specific license amendments where the review is not warranted.

8. Exceeding Performance or Repair Criteria

The proposed technical specifications require the licensee to monitor steam generator 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 [x.x.x]. Once shutdown, the licensee's steam generator program will ensure that the cause of the operational leakage is determined and corrective actions to prevent recurrence are taken. Operation may resume when the requirements of the licensee's program have been met. This requirement is unchanged from the existing Technical Specification.

The licensee may discover an error or omission during plant operation that indicates a failure to implement a required plugging or repair. Under these circumstances, the licensee is expected to take the actions required by their program. NRC notification will occur and the licensee will submit a report containing the cause, corrective actions to prevent recurrence, and the basis for the planned operating cycle. This requirement is new to the proposed technical specifications. The existing requirement addressed only operational leakage during operations.

During shutdown periods, the operational leakage criterion is no longer applicable, and the steam generators will be inservice inspected as required by the licensee's program. The licensee will perform a condition monitoring assessment of the "as found" condition of the steam generator tubes. The structural and accident leakage performance criteria are then used to assess the effectiveness of the licensee's program. This assessment may be performed analytically or by test. If the performance criteria are not met, the licensee will, through their steam generator program, ascertain the cause and determine corrective actions to prevent recurrence. Operation may resume when the requirements of the licensee's program have been met.

The proposed technical specifications do not change the actions required upon exceeding the operational leakage criterion.

The existing technical specifications do not address actions required while operating if it is discovered that a performance or repair criterion is exceeded, so the proposed change is an improvement.

If performance or repair criteria are exceeded while shutdown, required actions consist of repairing or plugging the affected tubes and reporting the condition to the NRC. The changes in the required reports are discussed under item 10 below.

9. Repair Methods

The proposed Technical Specifications require that tubes that exceed approved tube repair criteria be repaired in accordance with approved methods. For plants experiencing a damage form or mechanism for which no depth sizing capability exists, tubes identified with such damage are "repaired/plugged-on-detection" and integrity should be assessed. This requirement is unchanged from the existing Technical Specifications. However, the means of obtaining NRC approval of new repair methods is changed by this amendment. In addition, repair methods currently listed in the Technical Specifications are transferred to the [Technical Requirements Manual].

Repair methods are established in [Plants] steam generator program and, once approved, are listed in the [licensee's Technical Requirements Manual]. Tubes with defects exceeding the repair criteria shall be repaired with one of these approved methods.

The NRC Staff currently approves repair methods on a plant-by-plant basis. The Staff reviews each plant specific license amendment request and approves the proposal on a plant specific basis, including a plant specific technical specification change which, as a minimum, lists each method and may include specific technique requirements.

The proposed amendment will use the same approach for changing repair methods as was presented above for repair criteria. The NRC Staff would continue to have the opportunity to approve all repair methods that are implemented by domestic PWRs. However, the proposed approach would eliminate the wasteful expenditure of resources in the review of plant specific license amendments where the review is not warranted due to the availability of a generically approved document.

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

10. Reporting Requirements

The proposed amendment requires a report upon

- Failure to meet a performance criteria discovered during an inservice inspection,
- Failure to implement a required plugging or repair discovered while operating, and
- Greater than one percent of the tubes inspected in any one steam generator exceed the repair criteria.

The existing 15 day report and the 12 month annual reports are deleted. A Special Report is required if the inspection results in more than 1% of the tubes inspected per SG exceed the repair criteria.

The information included in the proposed reporting requirements envelopes that required by the existing Technical Specifications. In addition, the information provided is more useful in identifying the degradation mechanisms and determining their effects. Additional information, such as the basis for operating cycles, is required in the unlikely event that a performance criteria is not met.

Like much of the proposed amendment, 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 efficiently and, when required, more expeditiously than under the current Technical Specifications.

11. Definitions

The proposed [Technical Requirements Manual] introduces a number of new definitions that are important to the function of a steam generator program. The new definitions and their explanation are provided 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 exceed the offsite radiological dose consequences required by 10 CFR Part 100 guidelines, or the radiological consequences to control room personnel required by GDC-19.

2) ***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.*

   Since a burst definition is a required component for condition monitoring, a definition that can be analytically defined and is capable of being assessed via in situ and laboratory testing is required. Furthermore, the definition must be consistent with ASME Code definitions, and one that applies to most forms of tube degradation. Additionally, the definition is intended to demonstrate accord with the testimony of James Knight (Testimony of James Knight Before the Atomic Safety and Licensing Board, Docket Nos. 50-282 and 50-306, January 1975.), and compliance with the historical guidance of the Regulatory Guide 1.121. 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 safety factor in Section III of the ASME B&PV Code." Burst, or gross failure, according to the Code would be interpreted as a catastrophic failure of the pressure boundary.

   The proposed definition must also support 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, nor 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 definition is also intended not to characterize local instability, or for example, "ligament pop-through", as a burst. The onset of ligament tearing need not coincide with the onset of a full burst. As an example of not having a burst, consider an axial crack about 0.5" long with a uniform depth at 98% of the tube wall. Deformation during pressurization would be expected to lead to failure of the remaining ligament, (i.e., extension of the crack tip in the radial direction) 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 deep wear indications.

3) ***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 necessary.*

The definition of Normal Full Power Operation is important as it relates to application of the safety factor of 3 in the structural integrity performance criteria. The criterion requires "*...retaining a margin of 3.0 against burst for the pressure differential at normal steady state full power operation...*". The application of the safety factor of 3 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 margin 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.

4) ***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 existing technical specification through-wall (TW), depth-based criteria (i.e., 40% TW for most plants), a voltage-based repair limit per Generic Letter 95-05, or other alternative repair criteria (ARC). A steam generator degradation-specific management (SGDSM) strategy is followed to develop and implement an ARC.

Tubes identified with a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged-on-detection" and integrity is assessed. Note: "Plug-on-detection" is not considered an ARC.

An ARC methodology will be reviewed and approved by the NRC prior to its first time use at a licensed facility. Subsequent use of the same ARC at [Plant] will be justified by a safety evaluation that shows that [Plant] design falls within the parameters defined by the NRC in the SER approving the ARC.

5) ***Repair Methods*** *are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a steam generator tube is not a repair.*

The purpose of a repair is typically to reestablish or replace the reactor coolant pressure boundary. Repair methods are qualified and implemented in accordance with industry standards. The qualification of the repair techniques considers the specific steam generator conditions and mockup testing.

New repair methods will be reviewed and approved by the NRC prior to its first time use at a licensed facility. Subsequent use of the same method at [Plant] can be justified by a safety evaluation that shows that [Plant] design falls within the parameters defined by the NRC in the SER approving the repair method.

6) ***Steam generator tubing*** *refers to the entire length of the tube 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 steam generator tubes that are part of the RCS pressure boundary, 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 steam generator 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 are different.

E. Safety Analysis

The proposed amendment does not affect the design of the steam generators, their method of operation, or primary coolant chemistry controls. The proposed changes define limits that are at least as conservative as are currently in the plant licensing basis. The proposed Technical Specification change does not adversely impact any other previously evaluated design basis accident.

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 technical specifications plus the leakage rate associated with a double-ended rupture of a single tube is assumed.

The consequences of design basis accidents such as SGTR are, in part, functions of the dose equivalent I^{131} 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. For most PWRs, the SGTR accident is the limiting design basis event that establishes these technical specification limits.

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 the tubes will exhibit primary-to-secondary leakage that is at the operational leakage limit allowed by technical specifications.

The typical analysis of the above design basis accidents assumes that primary-to-secondary leakage for all steam generators 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 of dose equivalent I^{131} are at the technical specification values. For accidents that do involve fuel damage, the primary coolant activity values are a function of the accident conditions. None of these assumptions are affected by the proposed technical specification change.

The proposed technical specification change includes a reduction in the existing technical specification operational leakage limit. The limit of [150 gallons per day per steam generator], is based on operating experience as an indication of one or more tube leaks. This reduced leakage limit provides additional assurance that leaking flaws will not propagate to burst prior to plant shutdown.

In addition, the requirements proposed in this amendment are more effective in detecting steam generator degradation and prescribing corrective actions than are the existing Technical Specifications. As a result, the function and integrity of the tubes is maintained with greater assurance.

Therefore, the proposed change does not affect the consequences of a SGTR or any other design basis accident and the likelihood of such an accident is reduced.

F. Conclusions

The proposed license amendment will provide greater assurance of steam generator 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, and with the license change package developed by the NEI Steam Generator Task Force.

Adopting the changes proposed by this license amendment will provide added assurance that steam generator tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity.

Enclosure 6

Template for a License Amendment Significant Hazards Consideration

Template for No Significant Hazards Consideration

The proposed change has been evaluated against the standards in 10 CFR 50.92 and has been determined to not involve a significant hazards consideration, in that operation of the facility in accordance with the proposed amendments:

1. *Would not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The proposed change requires a steam generator program that includes performance criteria that will help ensure that the steam generator tubing will retain structural 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 steam generator performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

   The structural integrity performance criterion is:

   > *Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining margin of 3.0 against burst under normal steady state full power operation and a margin of 1.4 against burst under the most limiting design basis accident for structural integrity considerations, concurrent with a safe shutdown earthquake.*

   The structural integrity performance criterion is a new requirement. It is documented in the [Technical Requirements Manual].

   The accident-induced leakage performance criterion is:

   > *The primary to secondary accident induced leakage rate for the limiting design basis accident 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 steam generator, except for specific types of degradation at specific locations where the tubes are confined, as approved by the NRC and enumerated in conjunction with the list of approved repair criteria in the TRM].*

The accident induced leakage criterion is a new requirement. It is documented in the [Technical Requirements Manual].

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].*

The operational leakage criterion is a reduction of an existing requirement contained in the current technical specifications.

The steam generator performance criteria identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the steam generator 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.

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 technical specifications plus the leakage rate associated with a double-ended rupture of a single tube.

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). 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.

<u>Probability of an Accident</u>

The steam generator program referenced by this proposed amendment includes requirements that are a significant improvement over the requirements in the existing technical specifications. The steam generator program requirements affect many areas, including:

- Defining steam generator performance criteria

- Requiring a degradation assessment
- Requiring a condition monitoring assessment
- Reducing allowed operational leakage
- Requiring performance based inspections
- Establishing NDE requirements

As a result, the function and integrity of the tubes is maintained with greater assurance and the probability of a steam generator tube rupture is decreased.

Consequences of an Accident

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 the 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, and that the reactor coolant activity levels of dose equivalent I^{131} are at the technical specification values before the accident.

The operational leakage limit proposed by this technical specification amendment, [150 gallons per day per steam generator], establishes the acceptance limit for leakage existing prior to an accident. This limit is a [reduction in the value] allowed by the current technical specifications. The post accident (other than for a SGTR) leak rate limit remains at the value assumed by the accident analysis [(typically 1 gpm)]. Since the proposed operational leakage limit is more conservative than the existing value, it will not increase the likelihood or the consequences of an accident.

Conclusion

The proposed amendment does not affect the design of the steam generators, their method of operation, or primary coolant chemistry controls. The proposed Technical Specification change does not adversely impact any other previously evaluated design basis accident.

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 change does not affect the consequences of an MSLB, rod ejection, or a reactor coolant pump locked rotor event.

2. *Would not create the possibility of a new or different kind of accident from any other accident previously evaluated.*

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

Implementation of the proposed steam generator program will not introduce any adverse changes to a plant design basis or postulated accident resulting from potential tube degradation as a result of the implementation of the steam generator program is bounded by the existing tube rupture analysis. Primary-to-secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions.

The proposed amendment does not affect the design of the steam generators, their method of operation, or primary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component.

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

3. *Would not involve a significant reduction in a margin of safety.*

The steam generator (SG) tubes in pressurized water reactors 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 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 steam generator is maintained by ensuring the integrity of its tubes

Steam generator tube integrity is a function of the design, environment, and current physical condition. The proposed license amendment does not affect tube design or operating environment. The proposed changes are expected to result in an improvement in the tube integrity by implementing the steam generator program to manage steam generator 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 existing technical specifications.

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

Enclosure 7

Draft Revision 1 to
NEI 97-06
Steam Generator Program Guidelines

NEI 97-06

Revision 1B

Changes from Revision 0 are in bold blue font.

NEI 97-06 **[Rev 1B]**

Steam Generator Program Guidelines

January 2000

NEI 97-06 **[Rev 1B]**

Nuclear Energy Institute

Steam Generator Program Guidelines

January 2000

Nuclear Energy Institute, 1776 I Street N.W., Suite 400, Washington D.C. (202.739.8000)

ACKNOWLEDGMENTS

The Nuclear Energy Institute (NEI) Task Force on Steam Generator Programs developed the *Steam Generator Program Guideline* with oversight provided by the NEI Steam Generator Issues Working Group. We appreciate those industry contributors who reviewed and commented on this document to improve its technical content and its clarity.

NEI also wishes to thank EPRI. EPRI, through the Steam Generator Management Project, developed the steam generator guidelines referenced in this document.

NOTICE

EXECUTIVE SUMMARY

NEI 97-06 establishes a framework for structuring and strengthening existing steam generator programs. **It provides** the fundamental elements expected to be included in a steam generator program. These elements incorporate a balance of prevention, inspection, evaluation, repair and leakage monitoring measures.

This guideline refers licensees to EPRI guidelines for the detailed development of these programmatic attributes. EPRI will maintain these guidelines through the Steam Generator Management Project consensus process. Revisions to the EPRI documents will follow the protocol as noted in Section 1.5 of this document.

The intent of this document is to bring consistency in application of industry guidelines relative to managing steam generator programs. This document and those it references recognize the need for flexibility within each plant-specific program to adjust for the degree of degradation experienced and expected improvements in techniques for managing tube degradation.

Section 1, "Introduction", provides a background, discusses regulatory interface, **licensee** responsibilities, and protocol for revision of the referenced EPRI guidelines.

Section 2, "Performance Criteria", defines the performance criteria that **licensees** shall use to measure tube integrity. Meeting the performance criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its intended safety function of maintaining RCPB integrity.

Section 3, "Steam Generator Program", discusses the program elements and implementing guidance for strengthening existing steam generator programs.

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1. INTRODUCTION

1.1 PURPOSE

The purpose of this document is to bring consistency in application of industry guidelines related to managing steam generator programs. The framework offered in this document incorporates a balance of prevention, inspection, evaluation, repair and leakage monitoring measures. Additionally, this document establishes performance criteria that **licensees** shall use under the maintenance rule.

1.2 BACKGROUND

The program elements described in this document are evidence of the nuclear industry's commitment to safe and reliable steam generator operation. These elements focus on issues relative to the management and repair of steam generator tubing. For over two decades, the industry has expended considerable resources developing guidance on structuring steam generator programs to meet the challenges posed by tube degradation.

Chemistry control is an example of the industry's commitment to the resolution and management of steam generator degradation. By the mid-1970s, **licensees** were plugging tubes at a rate that would exceed steam generator 40-year-life design margins. The dominant damage form at that time was tube wastage. The industry corrected this by changing to an all-volatile water chemistry control. This, however, resulted in conditions conducive to corrosion of the carbon steel support plates, which led to tubing deformation as a result of denting and cracking with the same unacceptable rate of tube plugging. The industry, working through EPRI, met these challenges by implementing steam generator programs with aggressive improvements in control of secondary-side water chemistry and upgrades in secondary-side equipment, thus essentially eliminating both wastage and denting. The industry incorporated these successful programmatic strategies in the EPRI *Secondary Water Chemistry Guidelines* and associated supporting documents.

These chemistry guidelines have proven to be the cornerstones of the industry's effort to maintain acceptable steam generator performance. Over time, the industry's steam generator programs have matured to include improvements in programmatic features, such as non-destructive examination, primary-to-secondary leakage monitoring, and degradation-specific management. Building on the collective expertise of the industry, the EPRI Steam Generator Management Project (SGMP) oversees the maintenance of these guidelines, to incorporate technological and programmatic improvements.

1.3 Licensee Responsibilities

Revision 1 of NEI 97-06 was developed in conjunction with NRC interaction to support necessary Technical Specification changes to reflect enhanced steam generator monitoring. Revision 1 is to be implemented through a proposed technical specification amendment and associated documents. Each licensee shall ensure that existing regulatory requirements are met during implementation of NEI 97-06.

Each **licensee** shall adopt the performance criteria contained in Section 2. The performance criteria are (1) Structural Integrity, (2) Accident-Induced Leakage and (3) Operational Leakage. Further, each **licensee** shall evaluate existing program elements against those described in Section 3 and revise and strengthen, where necessary, to meet the intent of this document and the referenced EPRI guidelines.

The steam generator program **described in this document requires adherence to the intent of both the integrity elements and support elements discussed below. In addition, incorporation of the integrity elements is necessary to meet the requirements of the proposed technical specifications.**

The integrity and support elements are as follows:

Integrity Elements:
- **assessment of potential degradation mechanisms**
- **inspection**
- **integrity assessment**
- **maintenance and repairs**
- **primary-to-secondary leakage monitoring**
- **maintenance of secondary-side integrity**
- **reports to NRC**

Support Elements:
- **secondary-side water chemistry**
- **primary-side water chemistry**
- **foreign material exclusion**
- **self assessment**
- **reports to industry**

Section 3 provides additional information on these program elements.

1.4 Regulatory Requirements

The following section addresses NRC requirements that licensees should include in the development and implementation of the plant-specific steam generator program.

1.4.1 10 CFR Part 50 Appendix A, General Design Criteria for Nuclear Power Plants, and Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

General Design Criteria (GDC) 1, 2, 4, 14, 30, 31 and 32 of 10 CFR Part 50, Appendix A, define requirements for the reactor coolant pressure boundary (RCPB) with respect to structural and leakage integrity. Steam generator tubing and tube repairs constitute a major fraction of the RCPB 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.

General Design Criteria (GDC) 19 of 10 CFR Part 50, Appendix A, defines requirements for the control room and for the radiation protection of the operators working within it. Accidents involving the leakage or burst of steam generator tubing comprise a challenge to the habitability of the control room. Steam generator tubing and associated repair techniques and components, such as plugs and sleeves, must be capable of maintaining reactor coolant inventory and pressure in order to prevent excessive leakage and the resulting radiation doses to the control room operator.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction and operation of safety-related components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of these components; these include, in part, inspecting, testing, operating and maintaining. Criteria IX, XI, and XVI of Appendix B apply to the steam generator tube integrity program.

1.4.2 10 CFR § 50.65, *Maintenance Rule*

Under the maintenance rule, **licensees** classify steam generators as **risk significant** components because they are relied on to remain functional during and after design basis events. The performance criteria in Section 2 of this document shall be used to demonstrate that the condition of the steam generator "is being effectively controlled through the performance of appropriate preventive maintenance" (Maintenance Rule §(a)(2)). This guideline and the referenced EPRI guidelines define a steam generator program that provides the appropriate preventive maintenance that meets the intent of the maintenance rule.

Steam generators are to be monitored under §(a)(2) of the maintenance rule against industry-established performance criteria. If the performance criteria are not met, a cause determination of appropriate depth shall be done and the results evaluated to determine if goals should be established per §(a)(1) of the maintenance rule. NUMARC 93-01 [1] offers guidance for implementing the maintenance rule should a **licensee** elect to incorporate additional monitoring goals beyond the scope of this document.

1.4.3 10 CFR § 50.72, *Immediate Notification Requirements for Operating Nuclear Power Reactors,* and § 50.73, *Licensee Event Report System*

Failure to meet the performance criteria should be assessed to determine if it results in degradation of safety barriers. If so, the reporting requirements of §50.72 **and** §50.73 should be reviewed to determine applicability.

1.4.4 10 CFR § 100,

10 CFR § 100 establishes reactor siting criteria, particularly with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving the leakage or burst of steam generator tubing may comprise a challenge to containment and therefore involve an increased risk of radioactive release. Steam generator tubing and associated repair techniques and components, such as plugs and sleeves, must be capable of maintaining reactor coolant inventory and pressure in order to prevent excessive leakage.

1.4.5 Plant Technical Specifications

<u>Primary-to-Secondary Leakage</u> - **Plant technical specifications include a requirement to shut down when the plant exceeds an established threshold of primary-to-secondary leakage.**

<u>Steam Generator Program</u> - **Plant technical specifications require the development of a steam generator program, performance of a condition monitoring assessment of the steam generators, and place controls on the approval of new alternate repair criteria and tube repair methods.**

<u>Steam Generator Reporting Requirements</u> - **Plant technical specifications include requirements for steam generator reporting.**

1.5 PREPARATION AND REVISION PROTOCOL FOR EPRI GUIDELINES

The requirements in the **EPRI guidelines** represent a consensus of the committee and are experience-based in that they are achievable with available technology. Requirements will be incorporated into the EPRI guideline documents when it has been successfully demonstrated that the requirement can be applied in operating plants. **Meeting the intent of the EPRI Guidelines (References 2 through 7) is required.**

When a licensee's steam generator program deviates from the applicable guideline, a technical justification for deviation should be written and approved in accordance with the licensee's steam generator program. The technical

justification should provide the basis for the determination that the proposed deviation meets the same intent established by the applicable documents. Detailed guidance on justifying deviations is provided in Reference 11.

The responsibility for development and/or revision of EPRI Guidelines is typically assigned to the cognizant EPRI Steam Generator Management Project (SGMP) Issues Resolution Group (IRG) or the Technical Support Subcommittee (TSS).

Draft versions of documents or guidelines are typically generated as part of the interactive process of document development. Whenever possible and appropriate, it is desirable that these documents receive a "broad base" review and therefore the documents are normally distributed to the Technical Advisory Group (TAG) for review.

EPRI Guidelines are approved by the following groups in the order indicated:

1. **Guideline or Ad-Hoc Committee responsible for development**
2. **IRG or TSS assigned oversight responsibility for the document**
3. **Issues Integration Group (IIG)***
4. **Executive Group***

*** IIG and Executive Group approval is required for all the guidelines listed in sections 1.5.1 and 1.5.2 except for the *PWR Steam Generator Tube Plug Assessment Document*, [8]; and the *PWR Sleeving Assessment Document* [9].**

Additional guidance on the EPRI SGMP protocol is provided in Reference 11.

The **EPRI** guidelines referenced herein are:

- *PWR Steam Generator Examination Guidelines* [2];
- *PWR Primary-to-Secondary Leak Guidelines* [3];
- *PWR Secondary Water Chemistry Guidelines* [4];
- *PWR Primary Water Chemistry Guidelines* [5]:
- *Steam Generator Integrity Assessment Guidelines* [6];
- *In Situ Pressure Testing Guidelines* [7];

Additional information on plugging and repair can be found in the following assessment documents:

- *PWR Steam Generator Tube Plug Assessment Document*, [8]; and
- *PWR Sleeving Assessment Document* [9].

At an interval not to exceed two years, the EPRI Nuclear Power Council (NPC) will convene a utility committee(s) to review the applicable **EPRI guideline** to determine the need for revision.

Committee members include utility personnel, supplemented, as appropriate, by consultants, NSSS vendor and other supplier and/or service vendor personnel, all with equal voting rights. The members will have expertise relevant to the particular area being addressed. These committees are responsible to, and under the charter of, a utility sponsor group that broadly represents the management of the plants to which the prepared guidance is applicable. There will be an EPRI staff member on the committee, usually the chairperson, who will be a non-voting member. The NPC will approve the membership on the committees.

Once the committee prepares a final draft, it is circulated for broad industry review. The committee then resolves all comments generated as a result of the review and prepares a final document to be approved and issued by the sponsor group.

The NEI Steam Generator Review Board should be consulted **if questions arise on the interpretation of the EPRI guidelines. Such questions should be addressed and resolved before implementation in the steam generator program.**

When **NEI 97-06 is revised, licensees** will modify their steam generator **programs accordingly within 6 months.** If the next refueling outage is less than six months away, the **licensee** may delay incorporating appropriate changes **for an additional 3 months. When an EPRI Guideline is revised, licensees will modify their steam generator programs as directed by EPRI SGMP.**

2. PERFORMANCE CRITERIA

The steam generator (SG) tubes in pressurized water reactors have a number of important safety functions. These 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 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. The SG tubes are also relied upon to isolate the radioactive fission products in the primary coolant from the secondary system.

The steam generator performance criteria described below identify the standards against which performance is to be measured. Meeting the performance criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining RCPB integrity.

Performance criteria used for steam generators shall be based on tube structural integrity, accident-induced leakage, and operational leakage as defined below. **NRC approval is required for changes to the performance criteria.**

2.1 STRUCTURAL INTEGRITY PERFORMANCE CRITERION

The structural integrity performance criterion is:

> ***Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a margin of 3.0 against burst under normal steady state full power operation and a margin of 1.4 against burst under the limiting design basis accident concurrent with a safe shutdown earthquake.***

The structural performance criterion is based on ensuring that there is reasonable assurance that a steam generator tube will not **burst** during normal or postulated accident conditions. Section **3.1.3** of this guideline establishes the essential elements to meet this performance criterion.

The EPRI *Steam Generator Integrity Assessment Guideline* [6] offers guidance for the evaluation methods, **required margins and adjustments, and the typical inputs and assumptions used to determine tube integrity. It is important that the tube integrity assessments also account for input variability and uncertainties so as to provide a conservative assessment of the condition of the tubing relative to the performance criteria.**

In addition to the safety factor of three (3) for normal steady state operation and 1.4 for accident pressures, the integrity evaluation shall verify that the primary pressure stresses not exceed the yield strength for the full range of normal operating conditions as described in the performance criteria. Additionally, all appropriate loads contributing to combined primary plus secondary stress shall be evaluated so as to ensure that these loads do not significantly reduce the burst pressure for the full range of normal operating conditions including postulated accidents. For example, axial loads due to tube-to-shell temperature differences in once-through steam generator designs during postulated MSLB, or axial loading associated with locked tube supports in recirculating steam generator designs should be addressed to ensure that the types of degradation evaluated are not adversely impacted by these conditions.

2.2 Accident-Induced Leakage Performance Criterion

The accident-induced leakage performance criterion is:

> *The primary to secondary accident induced leakage rate for the limiting design basis accident, 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 steam generator, except for specific types of degradation at specific locations where the tubes are confined, as approved by the NRC and enumerated in conjunction with the list of approved repair criteria in the TRM].*

The pressure and temperature conditions used in the determination of the accident induced leakage rate shall be consistent with the conditions assumed in the accident analysis.

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 exceed the offsite radiological dose consequences required by 10 CFR Part 100 guidelines or the radiological consequences to control room personnel required by GDC-19.

In most cases when calculating offsite doses, the safety analysis for the limiting design basis accident assumes 1 gpm primary-to-secondary leakage as an initial condition. Plant specific assumptions for accident-induced leakage are defined in each licensee's licensing basis. Changes to the licensing basis require appropriate regulatory reviews.

Probabilistic safety analysis sensitivity studies have shown that severe accident risk is sensitive to certain design basis parameters such as 1 gpm accident induced leakage. Leakage rates greater than 1 gpm per steam generator could possibly cause failure in adjacent tubes under the conditions associated with severe accident scenarios. As a result, leakage greater than [Plant's] design basis or 1 gpm per steam generator is not allowed.

2.3 Operational Leakage Performance Criterion

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 should commence if primary-to-secondary leakage exceeds the Operational Leakage performance criterion. The measurement and detection methods, and associated actions shall meet the intent of the PWR Primary-to-Secondary Leak Guidelines [3]. Plant specific degradation mechanisms may exist which require a plant to implement reduced operational leakage limits.

The pressure and temperature conditions used in the determination of Operational Leakage shall be consistent with the guidance in the PWR Primary to Secondary Leak Guidelines [3].

3. STEAM GENERATOR PROGRAM

The purpose of a steam generator program is to ensure tube integrity. The program should contain a balance of prevention, inspection, evaluation and repair, and leakage monitoring measures. Licensees shall document the program through plant procedures. The major program elements are discussed below.

3.1 Steam Generator Integrity Elements

The guidance presented in section 3.1, Steam Generator Integrity Elements, is critical to a steam generator program. These elements are intended to ensure steam generator tube structural and leakage integrity are maintained. Note that observance of these elements is required to comply with NEI 97-06 and the proposed technical specifications.

3.1.1 Assessment of Degradation Mechanisms

Prior to planned steam generator inspections, licensees shall perform an assessment of existing degradation mechanisms. The assessment shall address the reactor coolant pressure boundary within the steam generator, e.g., plugs, sleeves, tubes and the components that support the pressure boundary, such as secondary-side components. The assessment shall consider operating experience from other similar steam generators. The assessment shall also consider engineering analysis of the degradation mechanisms.

The purpose of the assessment is to identify degradation mechanisms and for each mechanism identified:

- choose techniques to test for degradation based on the probability of detection and sizing capability;
- establish the number of tubes to be inspected;
- establish the structural limits; and
- establish the flaw growth rate or a plan to establish the flaw growth rate.

The identification of these parameters allows a **licensee** to establish the inspection or repair criterion before an outage. If a plant identifies a new degradation, or if the measured parameters change, such as growth rate, the plant may need to adjust analytical parameters during an inspection as the condition monitoring or operational assessment dictates.

The assessment of potential degradation mechanisms affects both the inspection and structural components of the program. The inspection component identifies the technique's capability, including detection probability, sizing capability, and measurement uncertainty. It will also identify the sampling strategy. The structural component applies the information gathered from the inspection with flaw growth rate projections to establish the repair limit and/or cycle length.

To conduct an effective inspection, the **licensee** should integrate the structural and inspection components. EPRI *Steam Generator Integrity Assessment Guidelines* [6] and EPRI *PWR Steam Generator Examination Guidelines* [2] provide guidance for assessment of potential degradation mechanisms.

3.1.2 Inspection

Each **licensee** shall plan inspections according to the expected tube degradation and follow the inspection guidelines contained in the latest revision of the EPRI *PWR Steam Generator Examination Guidelines* [2].

Some of the important features include:

- sampling using a performance-based approach **as supported by the degradation and integrity assessment,**

- obtaining the information necessary to develop **degradation**, condition monitoring and operational assessments,

- qualifying the inspection program by determining the accuracy and defining the elements for enhancing NDE system performance, including technique, analysis, field analysis feedback, human performance and process controls.

3.1.3 Tube Integrity Assessment

Licensees shall assess tube integrity after each steam generator inspection. **The assessment will include all degradation mechanisms known to exist in the steam generator being evaluated.** The purpose of the integrity assessment is to ensure that the performance criteria have been met for the previous operating period (i.e., condition monitoring), and will continue to be met for the next period (i.e., operational assessment). **These assessments shall account for all significant uncertainties such as to provide a conservative assessment of the condition of the tubing relative to the performance criteria. Potential significant sources of uncertainty include uncertainties associated with the projected limiting defect or indication size, material properties, and structural model. Conservative assumptions should be employed to account for uncertainties not directly treated in the assessment.** The EPRI *Steam Generator Integrity Assessment Guideline* [6] offers guidance for the evaluation methods, margins, and uncertainty considerations used to determine tube integrity.

The choice of an evaluation method to verify tube integrity will depend on the uncertainty surrounding the particular degradation being assessed which can be highly dependent on the availability of data. **Licensees** may use activities such as in-situ pressure testing or pulling tubes to supplement the tube integrity analysis. Reference 6 provides guidance as to when to conduct in situ pressure testing to address past operating period performance. The EPRI *In Situ Pressure Testing Guidelines* [7] provide guidance on screening criteria for candidate tube selection, as well as for test methods and testing parameters.

If a licensee determines that the structural integrity or accident leakage performance criteria have not been satisfied during the prior operating period, an evaluation of causal factors for failing to meet the criteria shall be performed **and corrective measures shall be taken.** In this event, the licensee is required to notify the NRC as discussed in Section **3.1.7.**

For an unscheduled inspection due to primary-to-secondary leakage, the tube integrity assessment need only address the degradation mechanism that caused the leak, provided the interval between scheduled inspections is not lengthened.

Normally, licensees shall complete **an operational** assessment for the next operating period within 90 days after startup. **If completion** of this assessment is not possible due to the complexity of the analysis within the 90 day period, a preliminary assessment is acceptable as an interim measure. There should be reasonable assurance that the performance criteria will not be exceeded prior to **completing** the final assessment.

Licensees shall establish tube repair **criteria** for each active degradation mechanism **known to exist in the steam generator being evaluated**. Tube repair criteria shall be either the existing technical specification through-wall (TW), depth-based criteria (i.e., 40% TW for most plants), a voltage-based repair limit per Generic Letter 95-05 [12], or other alternative repair criteria (ARC). If licensees choose to develop and implement an ARC, **they must follow the approval requirements contained in the licensee's technical specifications. Approved repair criteria are listed in the licensee controlled document that defines the actions required upon failure to meet a performance criterion**.

For plants experiencing a damage form or mechanism for which no depth sizing capability exists, tubes identified with such damage are "repaired/plugged-on-detection" and integrity should be assessed. Note: "**Repair**/plug-on-detection" is considered **a subset of the depth based criterion.**

If a risk based assessment is required, guidance may be found in Regulatory Guide 1.174 [13].

3.1.4 Maintenance, Plugging, and Repairs

Licensees shall qualify and implement **plugging and** repair methods in accordance with industry standards. The qualification of the **plugging and** repair techniques shall consider the specific steam generator conditions and mockup testing. The purpose of the **plugging and** repair is typically to remove degraded tubing from service, thereby redefining the reactor coolant pressure boundary.

Licensees shall clearly identify engineering prerequisites and plant conditions prior to performing the **plugging or** repair. Process controls shall be identified to ensure proper performance of the **plugging and** repair including the consideration of post maintenance testing. Additionally, licensees shall perform a **pre-service** inspection of the **plugging or** repair consistent with the latest revision of the EPRI *PWR Steam Generator Examination Guidelines* [2].

The EPRI *PWR Steam Generator Tube Plug Assessment Document* [8] and the EPRI *PWR Sleeving Assessment Document* [9] provide further guidance for maintenance and repair of tubing.

New repair methods shall be reviewed and approved by the NRC in accordance with the plant's technical specifications.

Approved repair methods are listed in the licensee controlled document that defines the actions required upon failure to meet a performance criterion.

3.1.5 Primary-to-Secondary Leakage Monitoring

Licensees shall establish primary-to-secondary leakage monitoring procedures in accordance with the **intent of the** EPRI *Primary-to-Secondary Leak Guidelines* [3] **and in accordance with the Operational Leakage criterion contained in section 2.3.**

Primary-to-secondary leakage monitoring is an important defense-in-depth measure that assists plant staff in monitoring overall tube integrity during operation. Monitoring gives operators information needed to safely respond to situations in which tube integrity becomes impaired and significant leakage or tube failure occurs. Additionally, operational leakage is an important tool for assessing the effectiveness of a steam generator program. **Plants should assess any observed operational leakage to determine if adjustments to the inspection program or integrity assessments are warranted.**

Appropriate training shall be provided for personnel who respond to primary-to-secondary leakage events.

3.1.6 Maintenance of Steam Generator Secondary-Side Integrity

Secondary-side steam generator components shall be monitored if their failure could prevent the steam generator from fulfilling its intended safety-related function. The monitoring shall include design reviews, an assessment of potential degradation mechanisms, industry experience for applicability, and inspections, as necessary, to ensure degradation of these components does not threaten tube structural and leakage integrity or the ability of the plant to achieve and maintain safe shutdown.

3.1.7. Reports to the NRC

Final implementation of a steam generator program requires that appropriate license technical specifications and other licensee controlled documents be developed to capture the elements of NEI 97-06. NRC reporting requirements are contained in both the technical specifications and the licensee controlled documents.

3.2 STEAM GENERATOR SUPPORT ELEMENTS

The guidance presented in section 3.2, Steam Generator Support Elements, is important to a steam generator program. These elements are important to provide for the long term viability of the steam generator. Note that observance of these elements is required to comply with NEI 97-06.

3.2.1 Secondary-Side Water Chemistry

Each **licensee** shall have procedures for monitoring and controlling secondary-side water chemistry to inhibit secondary-side corrosion-induced degradation in accordance with the EPRI *PWR Secondary Water Chemistry Guidelines* [4].

3.2.2 Primary-Side Water Chemistry

Each **licensee** shall have procedures for monitoring and controlling primary-side water chemistry to inhibit primary-side corrosion-induced degradation in accordance with the EPRI *PWR Primary Water Chemistry Guidelines* [5].

3.2.3 Foreign Material Exclusion

Each **licensee** shall have procedures to monitor for loose parts and control of foreign objects to inhibit fretting and wear degradation of the tubing. This program should include the attributes below.

3.2.3.1 Secondary-Side Visual Inspection

The program should define when such inspections are to be performed, the scope of inspection, and the inspection procedures and methodology to be used. Loose parts or foreign objects that are found should be removed from the steam generators, unless it is shown by evaluation that these objects will not cause unacceptable tube damage. This evaluation should be maintained as part of the inspection record. Tubes found to have visible damage should be inspected non-destructively and plugged or repaired if the repair criteria are exceeded.

3.2.3.2 Control and Monitoring of Foreign Objects and Loose Parts

The program should include procedures to preclude the introduction of foreign objects into either the primary or secondary side of the steam generator whenever it is opened (e.g., for inspections, repairs, and modifications).

Such procedures should include, as a minimum:

- detailed accountability for all tools and equipment used during an operation;
- appropriate controls and accountability for foreign objects such as eyeglasses and film badges;
- cleanliness requirements; and
- accountability for components and parts removed from the internals of major components (e.g., reassembly of cut and removed components).

Licensees should have alarm response procedures for the loose part monitoring system.

3.2.4 Self Assessment

Licensees shall perform self assessments regarding the steam generator management program. This review shall be performed by knowledgeable utility personnel or a contractor with independent experts selected by the **licensee** on a periodic basis. An INPO assessment can be used as an adjunct to the self assessment. The self assessment should identify areas for program improvement, along with program strengths. The assessment**, or a combination of assessments,** shall include all of the essential program elements described in Section 3 above.

3.2.5 Industry Reporting

Industry reports are necessary to share information on degradation mechanisms, NDE technique applications, operating experience, and other items. This experience is shared through the EPRI SGMP and various reports. Guidance on industry reporting is provided in the EPRI guidelines.

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APPENDIX A

References

1. NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, *.

2. *PWR Steam Generator Examination Guidelines*, EPRI Report TR-107569, *.

3. *PWR Primary-to-Secondary Leak Guidelines*, EPRI Report TR-104788, *.

4. *PWR Secondary Water Chemistry Guidelines*, EPRI Report TR-102134, *.

5. *PWR Primary Water Chemistry Guidelines*, EPRI Report TR-105714, *.

6. *Steam Generator Integrity Assessment Guideline*, EPRI Report TR-107621, *.

7. *In Situ Pressure Testing Guidelines*, EPRI Report TR-107620, *.

8. *PWR Steam Generator Tube Plug Assessment Document*, EPRI Report TR-**109495, ***.

9. *EPRI PWR Sleeving Assessment Document,* EPRI Report TR-105962, *.

10. **NUREG 0844, *NRC Integrated Program for the Resolution of Unresolved Safety Issues***
    ***EPRI A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity*, (September, 1988).**

11. ***EPRI Steam Generator Management Program Administrative Manual.***

12. ***Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,* GL 95-05, *(*August 3, 1999).**

13. ***An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,* Regulatory Guide 1.174, (July 1998).**

*** Latest revision approved per section 1.5**

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APPENDIX B

List of Definitions

The following definitions are provided to ensure a uniform understanding of terms used in this guideline.

Accident-induced Leakage

The primary-to-secondary leakage occurring during postulated accidents other than a steam generator tube rupture. This includes the primary-to-secondary leakage existing immediately prior to the accident plus additional primary-to-secondary leakage induced during the accident.

Burst

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.

Condition Monitoring

A comparison of the as-found inspection results against the performance criteria for structural integrity and accident leakage. Condition monitoring assessment is performed at the conclusion of each operating cycle.

Degradation-Specific Repair Criteria

Repair criteria developed for a specific degradation mechanism and/or location, e.g., a degradation specific repair criteria for ODSCC at tube support plates or for PWSCC at the tube sheet expansion.

Faulted

The state of the steam generator in which the secondary side has been depressurized due to a main steam line break such that protective system response such as main steam line isolation, reactor trip, safety injection, etc., has occurred.

Limiting Design Basis Accident

The accident that results in either the largest differential pressure for structural considerations or the largest dose for accident leakage considerations.

Normal Steady State Full Power Operation

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 necessary.

APPENDIX B (Cont'd)

Operational Assessment

Forward looking evaluation of the steam generator tube conditions that is used to predict that the structural integrity and accident leakage performance will be acceptable during the next cycle. The operational assessment needs to consider factors such as NDE uncertainty, indication growth, and degradation-specific repair limits.

Performance Criteria

Criteria to provide reasonable assurance that the steam generator tubing has adequate structural and leakage integrity such that it remains capable of sustaining the conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena.

Primary Stress

The primary stress with respect to the structural integrity performance criteria is the primary membrane stress produced by the internal differential pressure acting on the steam generator reactor coolant pressure boundary.

Probability of Detection (POD)

Probability of Detection (POD) is a measure of NDE performance and is defined as the likelihood that a NDE system will detect a flaw. POD may be expressed as a function of the severity of degradation. For this case, POD is typically calculated by comparing destructive examination results with the predictions of the eddy current inspection (found or missed). Alternatively, POD may be expressed as a fraction of the total population of flaws that would be detected by the NDE system (e.g., POD=0.6 per Generic Letter 95-05 [12]).

Repair Limit

Those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging. The repair limit will be determined by either subtracting margins for NDE uncertainty and growth from the structural limit or by conducting a probabilistic analysis.

Secondary Stress

The secondary stresses with respect to the structural integrity performance criteria are those stresses resulting from dynamic loads obtained from the modal analysis of the steam generator and its support structure. Major hydrodynamic and flow induced forces should be considered.

APPENDIX B (Cont'd)

Steam Generator Degradation-Specific Management (SGDSM)

The use of inspection and/or repair criteria developed for a specific degradation mechanism, e.g., outside diameter stress corrosion cracking at tube support plates.

Steam generator Tubing

Steam generator tubing refers to the entire length of the tube 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.

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APPENDIX C

List of Abbreviations and Acronyms

ARC	Alternate Repair Criteria
CFR	Code of Federal Regulations
GDC	General Design Criteria
GPD	Gallons Per Day
IIG	**EPRI SGMP Issues Integration Group**
IRG	**EPRI SGMP Issues Resolution Group**
INPO	Institute of Nuclear Power Operations
MSLB	Main Steam Line Break
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ODSCC	Outer Diameter Stress Corrosion Cracking
POD	Probability of Detection
PWR	Pressurized Water Reactor
PWSCC	Pressurized Water Stress Corrosion Cracking
RCPB	Reactor Coolant Pressure Boundary
SG	Steam Generator
SGDSM	Steam Generator Degradation Specific Management
SGMP	Steam Generator Management Project

APPENDIX C (Cont'd)

SGTR	Steam Generator Tube Rupture
TAG	**EPRI SGMP Technical Advisory Group**
TR	Technical Report
TSS	**EPRI SGMP Technical Support Subcommittee**