NEI

NUCLEAR ENERGY INSTITUTE

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December 11, 2000

Mr. Samuel J. Collins
Director, Office of Nuclear Reactor Regulation
Mail Stop O5-E7
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Revised Industry Steam Generator Program Generic License Change Package

PROJECT NUMBER : 689

Dear Mr. Collins:

On February 4, 2000 NEI submitted the industry Steam Generator Program Generic License Change Package for NRC review and endorsement. Industry events during the subsequent months have delayed the NRC review and presented an opportunity for the industry to improve the submittal. A summary of the major changes from the February submittal is provided in Enclosure 1. The resulting package (Enclosures 2 through 8), which supercedes the earlier version in its entirety, is enclosed for your endorsement.

Although the events of the last ten months have resulted in some changes to the industry steam generator program guidance, the fundamental principles remain sound. Equally as important, the program has demonstrated its resiliency. The steam generator program guidance is designed to accommodate new knowledge and experience, and that is precisely what is occurring. The enclosed Generic License Change Package includes changes that reflect recent experience. In addition, the appropriate underlying technical documents are currently being revised or supplemental guidance is being developed as necessary to reflect new information.

The industry and the NRC worked diligently to address the technical and licensing issues that ultimately resulted in the February 2000 version of the Steam Generator Program Generic License Change Package. Although these documents have been revised since the previous submittal, the differences do not represent a change in our position on the issues.

ADD: Robert Rothman
40 encls

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Revision 1 to NEI 97-06 (Enclosure 9) reflects the resolution of the issues addressed during the development of the Generic License Change Package. A draft copy of this document is included to provide background information so that the regulatory requirements can be understood within the context of the entire steam generator program. We are not seeking NRC approval of this document and will issue it to the industry shortly.

Once the NRC approves the enclosed Generic License Change Package, we expect that PWR licensees will use Enclosures 2 through 8 as templates in the preparation of plant-specific license amendment requests that meet the intent of the requirements in the generic package. NEI confirmed this expectation last March when all PWR licensees affirmed that they intended to adopt the proposed package once the NRC endorsed it. This affirmation was predicated on the assumption that the NRC review would not result in substantive changes to the proposed documents.

Implementation of the content of the Generic License Change Package represents an improvement in steam generator operation, inspection, and maintenance. From our perspective, the NRC and industry are fundamentally in agreement with the content of this package. For these reasons, the industry looks forward to a prompt response from the NRC in support of our initiative.

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."

Please contact Jim Riley (202-739-8137, jhr@nei.org) if you have any questions regarding these matters.

Sincerely,

David J. Modeen

David J. Modeen

JHR/maa
Enclosures:

1. SG Program Generic License Change Package Changes since February 4, 2000
2. Template for a Plant Specific License Amendment Cover Letter
3. Template for an Operational Leakage Technical Specification
4. Template for an Administrative Section Technical Specification
5. Template for a Licensee Controlled Document (Technical Requirements Manual)
6. Template for a Technical Requirements Manual Bases
7. Template for a License Amendment Safety Analysis
8. Template for a License Amendment Significant Hazards Consideration
9. 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 Regulator y Commission
Mr. Edmund J. Sullivan, U. S. Nuclear Regulatory Commission
Mr. Robert L. Rothman, U. S. Nuclear Regulatory Commission
Mr. Peter C. Wen, Jr., U. S. Nuclear Regulatory Commission

Enclosure 1

Major Changes
To the
SG Program Generic License Change Package
Since February 4, 2000

Steam Generator Program Generic License Change Package

The revised version of the Steam Generator Generic License Change Package is provided in Enclosures 2 through 8. The most significant changes to the documents relative to the February version are summarized bellow:

Template for a License Amendment Cover Letter

The license amendment cover letter template is Enclosure 2 to this letter. The only significant change made to this document was a clarification - the TRM and TRM Bases are being submitted as part of the Generic License Change Package for information only. These documents are licensee controlled and are therefore not subject to NRC approval. The 10 CFR 50.59 process governs changes to these documents.

RCS Operational Leakage Tech Spec and Bases

The Operational Leakage Tech Spec and Bases are Enclosure 3 to this letter. The following significant changes were made to these documents

- Actions required upon exceeding RCS leakage limits were changed to remove the four-hour period provided for reducing primary-to-secondary leakage before commencing a shutdown. The four hour period was never included in the governing industry guidance.
- The relationship between the primary-to-secondary leakage limit in the Operational Leakage Technical Specification (150 gpd) and the accident analysis assumption (nominally 1 gpm) was clarified in the Bases.
- The Bases includes an explanation of the limitations of primary-to-secondary leakage measurements during low power conditions.

Administrative Technical Specification for a Steam Generative Program

This document (Enclosure 4) was not changed substantively. The format and some of the phrasing was revised to conform to standard Tech Spec convention and to clarify the intent.

Technical Requirements Manual

This document (Enclosure 5) was completely revised and significantly simplified, although its technical intent was not altered. An explanation of the major changes follows.

- THE LCO statement now includes the words: "...by plugging or repairing tubes that exceed the Repair Criteria during SG inspections." This change was made to emphasize the intention of the LCO – satisfactory completion of a Condition Monitoring / Operational Assessment (CM/OA) during a SG inspection is sufficient to ensure conformance with the Performance Criteria during the subsequent operating cycle unless an error in the integrity assessment is discovered. The change also makes the LCO statement consistent with the terminology in Condition A and thereby enhances operator understanding of the requirements.

- The structural integrity performance criterion was revised to remove the phrase "concurrent with a safe shutdown earthquake" since this condition is not within the licensing basis of some plants. In its place the following sentence was added: "Any additional loading combinations shall be included as required by existing design and licensing basis." For those plants whose licensing basis includes a safe shutdown earthquake, the performance criterion is essentially unchanged. It was never the intent of the criterion to impose requirements beyond a plant's licensing basis.

- The Accident Induced Leakage Performance Criterion was editorially revised – no substantive changes are intended.

- The Operational Leakage Performance Criterion was removed to avoid duplication since it was already included in the Operational Leakage Tech Spec.

- The Contingency Measure for failure to meet the Repair Criteria in MODES 5 and 6 has been removed. This requirement had created confusion by specifying actions that were in response to conditions that were not defined by the LCO, and that occurred during MODES in which the LCO was not applicable. The current version of the TRM accomplishes the same thing through the surveillance that requires tube plugging or repair prior to entry into MODE 4. This change also removed the possibility of interpreting the TRM to imply that SG operability was dependent on meeting the repair criteria.

- The Contingency Measure for failure to meet the Performance Criteria in MODES 5 and 6 has been removed. This requirement had created confusion by specifying actions that were in response to conditions that were not defined by the LCO, and that occurred during MODES in which the LCO was not applicable. The current version of the TRM accomplishes the same thing through the surveillance that requires verification of tube integrity prior to entry into MODE 4.

- An applicability statement was added to the TRM for MODES 1 through 4. SG operability is only tied to the Performance Criteria in MODES 1 through 4 and this was the intent of the February version where MODE applicability was expressed as part of the LCO statement.

- All the NRC reports were moved from the TRM LCO to the TRM Administrative Section. This is consistent with normal TRM standards and satisfied a number of industry comments concerning the way the TRM dealt with the NRC report requirements; it is not necessary to specifically include reports in the TRM if they are already required by regulation. No changes in the actual report requirements are intended.

- Surveillance frequency was tied to entry into MODE 4. This makes the frequency easier to understand while expressing the need to verify SG tube integrity prior to start up. The intent is not to review the previous Condition Monitoring and Operational Assessment, but rather to ensure that they are current and that there are no known problems that would cast doubt on their accuracy. This intent is explained in the TRM Bases.

- The definitions were moved from the TRM to the TRM Bases. They were also presented separately in the event that a licensee prefers to include them in some other licensee controlled document. The TRM typically has no definition section; the definitions in the TRM are usually the same as those in the Tech Specs. The Bases also includes the lists of approved Repair Criteria and Repair Methods that were previously included in the TRM definitions.

- The definition of Limiting Design Basis Accident was revised to clarify its intent within the context of the SG Program.

- The definition of Steam Generator Tubing was revised to clarify that tubing refers to the tube wall and any repairs made to the tube wall.

TRM Bases

The TRM Bases (Enclosure 6) was not included in the February version of the SG Program Generic License Change Package. It was created to facilitate understanding of the TRM in support of its implementation at license's plants.

Safety Analysis and Significant Hazards Consideration

The Safety Analysis and Significant Hazards Consideration (Enclosures 7 and 8) were changed to be consistent with the rest of the revised documents.

NEI 97-06 Draft Revision 1

Draft revision 1 to NEI 97-06 is provided in Enclosure 9. The most significant changes are summarized below:

- All references to the proposed Technical Specifications have been removed and the actual requirements inserted. The new Tech Specs will not be in place when revision 1 is issued.
- NEI 97-06 was revised throughout to maintain consistency with the SG Program Generic License Change Package.
- Guidance for justifying deviations from NEI 97-06 and its referenced EPRI Guidelines was added to NEI 97-06 as Appendix D.
- Section 1.5 now requires that revisions to NEI 97-06 and its referenced EPRI Guidelines include a summary of the sections changed along with a technical justification for the changes. This improves understanding of the changes and facilitates implementation.
- The intent of the industry reporting requirements in section 3.2.5 was clarified.

Enclosure 2

Template for a Plant Specific License Amendment Cover Letter

Revision to NRC Submittal

Changes with respect to the February 4, 2000 version are bold, underlined, blue font.

[Month Day, 2001]

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 Tube Surveillance 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. 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 [Technical Requirements Manual] is provided for information only to illustrate how the Steam Generator Program requirements are incorporated into plant documentation. [Plant] is not requesting NRC approval of the [Technical Requirements Manual].

Technical Specifications [3.4.13, 5.5.9, and 5.6.10] are revised as described below.

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 proposed amendment also deletes the existing LCO 3.4.13.d since it is enveloped by the revised LCO and revises the Conditions and Surveillances to clarify the requirements related to primary-to-secondary leakage.]

The changes to Administrative Technical Specification [5.5.9], Steam Generator Tube Surveillance Program, require the implementation of a Steam Generator Program that contains the steam generator performance criteria. The change removes the detailed inspection requirements from the Technical Specifications and replaces them with the essential elements of a program. 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 the intent of 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 [MM/YY] in order to allow implementation of the associated requirements for scheduled refueling outages after [MM/YY].

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 s 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 T ruly Yours,

Enclosures: ( ) Summary Description and Safety Analysis

( ) Determination of Significant Hazards
( ) Technical Specification Marked-up Pages

Enclosure 3

Template for an Operational Leakage Technical Specification And Bases

Revision to NRC Submittal

Changes with respect to the February 4, 2000 version are bold, underlined, blue font.

Insert A

APPLICABLE SAFETY ANALYSIS:

That primary to secondary LEAKAGE from all steam generators is [one gallon per minute] or increases to [one gallon per minute] as a result of accident induced conditions. The Technical Specification LCO requirement to limit primary to secondary leakage through any one steam generator to less than 150 gallons per day is significantly less than the condition assumed in the safety analysis.

Insert B

Primary to Secondary Leakage through any one SG

The limit of 150 gallons per day per steam generator is based on operating experience gained from SG tube degradation mechanisms which result in tube leakage. This leakage rate along with the other performance criteria in the Steam Generator Program provide reasonable assurance that a single flaw leaking this amount will not propagate to an SGTR under the stress conditions of a LOCA or a main steam line rupture prior to detection by leakage monitoring methods and commencement of plant shutdown. If leaked through many flaws, the flaws are very small and the above assumption is conservative.

The leakage rate limit applies to leakage in any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the leakage should be conservatively assumed to be from one steam generator. [However, plants with N-16 monitors on individual steam lines can quantify leakage in any one steam generator.]

The RCS Operational primary to secondary leakage is measured at standard temperature and pressure.

Insert C

SR 3.4.13.1:

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 an RCS water inventory balance. In addition, primary-to-secondary leakage may need to be checked more frequently than every 72 hours in order to ensure adequate protection against rapidly increasing SG tube leaks.

Insert D **SR3.4.13.2:**

This SR requires that the Operational Leakage performance criterion in the Steam Generator Program be satisfied. Satisfying the Operational Leakage performance criterion ensures that the primary to secondary LEAKAGE limit is met. The Operational Leakage performance criterion along with the other performance criteria in the Steam Generator Program provide reasonable assurance against tube burst at normal and faulted conditions. The 150 gallons per day limit is measured at standard temperature and pressure.

Primary to secondary leakage is determined through the analysis of secondary coolant activity levels. At low power, primary and secondary coolant activity is sufficiently low that an accurate determination of primary to secondary leakage may be difficult. Immediately after shutdown, the short lived isotopes are usually at sufficient levels to monitor for leakage by normal power operational means as long as other plant conditions allow the measurement. During startup, especially after a long outage, there are no short lived isotopes in either the primary or secondary system. This limits measurement of the leakage to chemical or long lived radiochemical means. The Steam Generator Program provides guidance on leak rate monitoring during MODES 3 and 4.

Enclosure 4

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

Revision to NRC Submittal

Changes with respect to the February 4, 2000 version are bold, underlined, blue font.

5.5.9 Steam Generator Program

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. **The Program shall include:**

a. 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.

b. Performance Criteria - The steam generator performance criteria are defined in the Steam Generator Program. Revisions to performance criteria (and their associated definitions as used in the Steam Generator Program) require review and approval by the NRC. The performance criteria (and their associated definitions as used in the Steam Generator Program) may be revised to incorporate changes approved generically by the NRC subject to the limitations and conditions set forth in the staff's approving document. Demonstration of satisfaction of the generic limitations and conditions must be documented in an evaluation prepared in accordance with 10 CFR 50.59.

c. Tube Repair Criteria and Repair Methods - Tube repair criteria and repair methods shall be described in and implemented by the Steam Generator Program. Repair criteria and repair methods may be implemented after review and approval by the NRC. In addition, repair criteria and repair methods approved generically by the NRC may be used subject to the limitations and conditions set forth in the staff's approving document. Demonstration of satisfaction of the generic limitations and conditions must be documented in an 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, a Special Report shall be submitted within 120 days after the initial entry into Mode 4 following completion of the inspection. The report shall summarize:

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

Enclosure 5

Template for a Licensee Controlled Document [Technical Requirements Manual]

Revision to NRC Submittal

Changes with respect to the February 4, 2000 version are bold, underlined, blue font.

Template for Steam Generator Tube Integrity Licensee Controlled Document

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

LCO Steam generator tubes shall meet primary to secondary pressure boundary integrity Performance Criteria as specified below by plugging or repairing tubes that exceed the Repair Criteria during SG inspections.

A. 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 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 Limiting Design Basis Accident. **Any additional loading combinations shall be included as required byexisting design and licensing basis.**

(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 NRC has approved greater accident-induced leakage as part of a plant's licensing basis. Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria].

(iii) Operational leakage criterion:

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

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

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

[CONTINGENCY MEASURES:]

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Failure to implement a required plugging or repair.	A.1 Determine steam generators remain acceptable for continued operation based on meeting the Performance Criteria.	In accordance with [the licensee's Corrective Action Program]
B. Required Action A.1 and associated Completion Time not met .	B.1 Declare the SG inoperable and enter the appropriate Conditions and Required Actions in the TS.	Immediately

[VERIFICATION REQUIREMENTS]

VERIFICATION		FREQUENCY
SR A.	Verify steam generator tube integrity is in accordance with the Performance Criteria described in the Steam Generator Program.	Prior to entry into MODE 4
SR B.	Verify that SG tubes that exceed the Repair Criteria are plugged or repaired in accordance with Repair Methods.	Prior to entry into MODE 4

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

The following reporting requirements must be included in an appropriate section of the licensee controlled document

6.x Steam Generator Performance and Repair Criteria Reports

6.X.1 Upon discovery of a failure to implement a required plugging or repair, the NRC shall be notified within 8 hours. A special report shall subsequently be submitted to the NRC within 60 days of the discovery that provides the basis for the planned operating period.

6.X.2 If a steam generator tube(s) is found to not meet the Performance Criteria during an outage that includes a steam generator inspection, the NRC shall be notified within 8 hours. A special report shall subsequently be submitted to the NRC within 60 days of the discovery that provides information on the Performance Criteria exceeded. Another special report that provides the basis for the planned operating period shall subsequently be submitted to the NRC within 120 days of entering MODE 4.

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

The following definitions apply to terms used within the context of the licensee controlled document and its Bases. They should be considered for inclusion in the implementing plant document.

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.

Limiting Design Basis Accident

In the context of steam generator primary-to-secondary pressure boundary integrity, it is the accident that results in either the largest differential pressure across the steam generator tubes for structural considerations or the minimum margin to the applicable dose limits 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.

Repair Criteria

Those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging.

Repair Methods

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

Steam Generator Tubing

Steam generator tubing refers to the entire length of the tube, including the tube wall and any repairs to it, between the tube-to-tube sheet weld at the tube inlet and the tube-to-tube sheet weld at the tube outlet. The tube-to-tube sheet weld is not considered part of the tube.

Enclosure 6

Steam Generator Tube Integrity
Technical Requirements Manual Bases

The TRM Bases was not submitted in February 2000, therefore no changes are indicated.

TRM Steam Generator Integrity

BASES

BACKGROUND

The purpose of the steam generator integrity LCO is to require compliance with the steam generator Performance Criteria. The steam generator Performance Criteria define the basis for steam generator OPERABILITY. The Performance Criteria apply to steam generator tubes and associated appurtenances (e.g. plugs, sleeves, and other repairs). The Performance Criteri a and the processes required to meet them are addressed by the Steam Generator Program.

Satisfying the Performance Criteria provides reasonable assurance of tube integrity at normal and faulted conditions. Steam generator tube integrity means that the tubes are capable of performing their intended safety functions consistent with their licensing basis, including applicable regulatory requirements.

Since the Performance Criteria can not be surveilled during MODES 1 through 4, the LCO defines conformance with the Performance Criteria by requiring that tubes that exceed the repair criteria are plugged or repaired during SG inspections.

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 isolate the radioactive fission products in the primary coolant from the secondary system.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. Steam generator tubes have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively.

BASES (continued)

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 remains capable of fulfilling its specific safety function of maintaining RCPB integrity.

APPLICABLE SAFETY ANALYSIS

Satisfying the Performance Criteria provides reasonable assurance against tube Burst and the resulting primary to secondary leakage that might occur at normal and faulted conditions. The consequences of design basis accidents that include primary to secondary leakage are, in part, functions of the accident induced primary-to-secondary leakage rates and the dose equivalent I^{131} in the primary coolant.

The typical analysis for an event resulting in steam discharge to the atmosphere, except a steam generator tube rupture (SGTR), 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 consistent with the technical specification values. For accidents that do involve fuel damage, the primary coolant activity values are a function of the accident conditions.

For most PWRs, the SGTR accident is the limiting design basis event that establishes limits for these parameters. 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 accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is steamed to the main condenser.

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 leakage is assumed to be at the design

BASES (continued)

basis value, which is consistent with the accident induced leakage performance criterion. Due to the large amount of fluid that is released during a MSLB, it is usually more limiting for site radiation releases.

The steam generator Performance Criteria in this document and the limits included in the plant technical specifications for operational leakage and for dose equivalent I^{131} in primary coolant ensure the plant is operated within its analyzed condition. The dose consequences resulting from the most Limiting Design Basis Accident are within the limits defined in GDC 19 [1], 10 CFR 100 [2] or the NRC approved licensing basis (i.e., a small fraction of these limits).

LCO

The LCO requires that steam generator Performance Criteria be met. Since conformance with the Performance Criteria can only be determined during SG inspections, the LCO is met during MODES 1 through 4 by verifying that tubes that exceed the repair criteria have been plugged or repaired.

The steam generator Performance Criteria include design basis parameters that define acceptable steam generator performance. Steam generator OPERABILITY is based on meeting the Performance Criteria.

Changes to the Performance Criteria are controlled by the license Technical Specifications.

A. Performance Criteria

The SG Performance Criteria are based on tube structural integrity, accident-induced leakage, and operational leakage. Compliance with the structural integrity and accident induced leakage performance criteria can only be determined during steam generator inspections. These inspections are performed when the reactor is shutdown and depressurized so that the primary side of the SG can be drained.

(i) Structural Criterion

The Structural 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 Limiting Design Basis Accident. Any additional loading combinations shall be included as required by existing design and licensing basis."

Steam Generator Tubing refers to the entire length of the tube, including the tube wall and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

In the context of the Structural Criterion for steam generator primary-to-secondary pressure boundary integrity, the Limiting Design Basis Accident is defined as the accident that results in the largest differential pressure across the steam generator tubes.

The Structural Criterion can be broken into two separate considerations:

- Providing a margin of safety against tube Burst under normal and accident conditions, and
- Ensuring structural integrity (preventing yield or Burst) of the SG tubes under all anticipated transients included in the design specification.

Tube Burst

Tube Burst is defined as the gross structural failure of the tube wall. The condition typically corresponds to an

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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 .

The Structural Criterion provides reasonable assurance that a steam generator tube will not Burst during normal or postulated accident conditions. The Structural Criterion requires that the tubes not Burst when subjected to differential pressures equal to three (3) times those experienced during normal steady state operation and 1.4 times accident differential pressures. In addition, other loading combinations are included as required by the design and licensing basis. The safety factors of 3 and 1.4 and the requirement to include applicable design basis loads are based on ASME Code Section III subsection NB [7] requirements and Draft Regulatory Guide 1.121 [8] guidance.

For most plants the Normal Steady State Full Power Operation condition defines the most limiting parameters under which the tubes are tested. In the context of the Structural Criterion, 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 their effects on differential pressure should be accounted for if significant. Guidance on accounting for changes in these parameters is provided in the EPRI Integrity Assessment Guidelines [6].

Tube Yield

The Structural Criterion verifies that the primary pressure stresses do not exceed the yield strength for the full range of normal operating conditions including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification. All appropriate loads contributing to combined primary plus secondary stress are 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 are addressed to ensure that the test conditions are at least as severe as those expected during operating and accident events.

(ii) Accident Induced Leakage Criterion

The Accident Induced Leakage 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 NRC has approved greater accident-induced leakage as part of a plant's licensing basis . Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria]."

In the context of the Accident Induced Leakage Criterion:

- 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.

- For steam generator primary-to-secondary pressure boundary integrity considerations, Limiting Design Basis Accident is defined as the accident that results in the minimum margin to the applicable dose limits.

The Accident Induced Leakage Criterion can be broken down into two separate considerations:

- Meeting design basis conditions, and
- Limiting accident induced leakage to less than 1 gpm per steam generator under all circumstances.

Design Basis

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a Limiting Design Basis Accident. The radiological dose consequences resulting from a potential primary-to-secondary leak during postulated design basis accidents must not exceed the offsite dose limits required by 10 CFR Part 100 [2] or the control room personnel dose limits required by GDC-19 [1] or the NRC approved licensing basis.

In most cases when calculating offsite doses, the safety analysis for the Limiting Design Basis Accident, other than a steam generator tube rupture, assumes a total of [1 gpm] primary to secondary leakage as an initial condition. Plant specific assumptions for accident induced leakage are defined in each licensee's licensing basis. The leakage value used in the Accident Induced Leakage Criterion must be consistent with the licensing basis.

Limiting Accident Induced Leakage to 1 gpm per SG

The NRC has stated that probabilistic safety analysis sensitivity studies have shown that accident risk is sensitive to certain design basis parameters such as 1 gpm accident induced leakage per SG. As a result, leakage greater than the design basis or 1 gpm per steam generator is not allowed unless the NRC has approved greater leakage rates as part of an Alternate Repair Criterion.

(iii) Operational Leakage Criterion

The Operational Leakage Criterion and its associated action and surveillance requirements are contained in the RCS Operational Leakage Technical Specification. The Operational Leakage Criterion is not included in the TRM to avoid duplication with the license Technical Specifications. The following summary of the Operational Leakage requirements is provided to facilitate an understanding of all of the Performance Criteria since they act together to define SG OPERABILITY.

The Operational Leakage 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) from any one steam generator. Operational leakage is measured at standard temperature and pressure conditions (freezing temperature of water, 0 degrees C, and atmospheric pressure, 14.7 psia).

The limit of 150 gallons per day per steam generator (the plant limit is 150 gpd times the number of SGs in the unit) is based on operating experience gained from SG tube degradation mechanisms which result in tube leakage. This leakage rate along with the other Performance Criteria in the Steam Generator Program provide reasonable assurance that

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a single flaw leaking this amount will not propagate to an SGTR under the stress conditions of a LOCA or a main steam line rupture prior to detection by leakage monitoring methods and commencement of plant shutdown. If the leakage occurs through many flaws, the flaws are very small and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is required during all MODES; however when the plant is shutdown in MODES 5 and 6 RCS conditions are far less challenging than during MODES 1 through 4, resulting in lower stresses and reduced potential for leakage. Therefore this LCO is applicable in MODES 1 through 4 only.

Steam generator tubes are designed to withstand the stresses due to differential pressures as large as 3 times those experienced under normal full power operations or 1.4 times those experienced during a Limiting Design Basis Accident. This requirement is delineated in the Structural Criterion. This magnitude of differential pressure is only possible during MODES 1, 2, 3, and 4. When the reactor is in MODES 5 and 6, primary to secondary differential pressure and primary coolant activity are so low that primary to secondary leakage is not a significant safety concern. During MODES 5 and 6 the Performance Criteria are still required to be met, but do not form a basis for SG OPERABILITY.

ACTIONS

A. Implementation of Repair Criteria can only be accomplished during SG inspections that are performed when the reactor plant is shutdown. If any plugging or repair of tubing is required, it is completed before the plant enters an operating MODE. If a required plugging or repair was not implemented during the last inspection, the SGs were returned to service with a tube already exceeding the Repair Criteria and the acceptability of the steam generators for continued operation is in question.

In order to determine SG OPERABILITY, an evaluation must be completed that starts with the physical condition of the tube at the time of its last inspection and accounts for the time since the inspection, and the potential growth rate of the degradation. The

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OPERABILITY determination is based on meeting the Performance Criteria using the estimated condition of the tube at the time the situation is discovered. A tube's failure to meet a Repair Criterion does not necessarily render the SG inoperable. Steam generator OPERABILITY is based on meeting the SG Performance Criteria since the Performance Criteria provide reasonable assurance that the SG tubing remains capable of meeting its safety function. The SG Repair Criteria define limits on SG tube degradation that allow for flaw growth between inspections and still provide assurance that the Performance Criteria will continue to be met.

The plant's Corrective Action Program will establish the Completion Time requirement for the OPERABILITY determination.

If an operating plant discovers that required plugging or repair was not implemented during a previous steam generator inspection, the NRC must be notified. These reporting requirements are specified in [section 6.0, Reporting Requirements, of the TRM]. Two reports are required.

Initially the NRC must be notified within 8 hours. This notification is based on the requirement in 10 CFR 50.72 (b) (3) (ii) (A) [3] to notify the NRC within 8 hours of a serious degradation of a plant's principal safety barriers.

In addition, a special report is submitted within 60 days of determining the failure to implement a required plugging or repair. This report is intended to be coincident with the report required by 10 CFR 50.73 (a) (2) (ii) (A) (serious degradation of the principal safety barriers) [4]. In addition to the content required by the regulation, the report must provide information on the specific Performance Criteria exceeded and the basis for the next operating period (Operational Assessment). This determination is important in order to provide reasonable assurance that the Performance Criteria are currently being met and will continue to be met at the time of the next SG inspection. The 60 day requirement is based on NEI 97-06 [5] that requires completion of a tube integrity assessment for the next operating

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cycle within 90 days after startup. The Action requirement for the report is more restrictive since this report is necessary to prove that the degraded SG tube(s) are currently meeting the Performance Criteria.

B. If the OPERABILITY evaluation required by Action A. 1 determines that a SG tube does not meet a Performance Criteria in MODES 1 through 4, the steam generator must be declared inoperable and the actions required by the appropriate license Technical Specifications must be followed. The steam generator Performance Criteria include the design basis parameters that define acceptable steam generator performance. Meeting the Performance Criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining RCPB integrity. Therefore, OPERABILITY of the steam generators is based on meeting the Performance Criteria.

SURVEILLANCE REQUIREMENTS

A. During shutdown periods the steam generators will be inspected as required by the licensee's Steam Generator 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 tube integrity and the effectiveness of the Steam Generator program. This assessment may be performed analytically or by test.

The existence of the Steam Generator Program is required by the Technical Specifications. NEI 97-06, Steam Generator Program Guidelines [5], and its referenced EPRI Guidelines establish the content of the Steam Generator Program.

The Steam Generator Program requires periodic steam generator tube integrity assessments. 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). The condition monitoring assessment

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is an evaluation of the "as found" condition of the tubing with respect to the Performance Criteria for structural and accident leakage integrity. The operational assessment determines the length of the operating cycle by providing reasonable assurance that the tubing will meet the Performance Criteria at the next scheduled inspection.

Tube integrity must be verified prior to entry into MODE 4 because the tubes must meet the Performance Criteria in MODE 4 in order for the steam generators to be OPERABLE. The surveillance requirement can be satisfied by determining that an acceptable Condition Monitoring / Operational Assessment exists and that the plant is within the operating cycle determined by the Operational Assessment.

If a plant discovers that a Performance Criteria has not been met during a steam generator inspection, the NRC must be notified. The reporting requirements are specified in [section 6.0, Reporting Requirements , of the TRM]. Three reports are required.

Initially the NRC must be notified within 8 hours. This notification is based on the requirement in 10 CFR 50.72 (b) (3) (ii) (A) [3] to notify the NRC within 8 hours of a serious degradation of the principal safety barriers.

In addition, a special report is required within 60 days of determining a Performance Criterion was not met. This report is intended to be coincident with the report required by 10 CFR 50.73 (a) (2) (ii) (A) (serious degradation of the principal safety barriers) [4]. In addition to the content required by the regulation, the report must provide information on the specific Performance Criteria exceeded. The 60 day time limit is required by 10 CFR 50.73 [4].

Finally, a special report is required within 120 days of determining a Performance Criterion was not met. This report must describe the basis for the next operating period (Operational Assessment). This determination is important in order to provide reasonable assurance that the Performance

BASES (continued)

Criteria will be met at the time of the next SG inspection. The 120 day completion time is based on an NEI 97-06 [5] requirement that specifies completion of a tube integrity assessment for the next operating cycle within 90 days after startup.

B. During steam generator inspections the licensee wil l perform an integrity assessment of the steam generator tubes. 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).

Ensuring that the tubes will meet the Performance Criteria at the next inspection involves the use of Repair Criteria. Repair Criteria are those NDE measured parameters at or beyond which a tube must be repaired using an approved Repair Method or removed from service by plugging. The tube Repair Criteria establish limits for tube degradation that provide reasonable assurance that an affected tube will meet the Performance Criteria at the next scheduled inspection by allowing for anticipated growth during the intervening time interval. Because of this allowance for growth, exceeding a tube repair criterion does not necessarily render the steam generators inoperable.

Tube Repair Criteria are either the existing technical specification through-wall (TW) depth-based criterion (i.e., 40% TW for most plants), or other Alternate Repair Criteria (ARC) approved by the NRC such as a voltage-based repair limit per Generic Letter 95-05.

The depth base criterion, approved for use at all plants by the NRC, was established when the most frequent form of degradation was general wastage corrosion. This type of degradation structurally bounds other forms of degradation and is characterized by a volumetric loss of the tube wall. This criterion was established to allow for NDE uncertainties and growth and still provide a reasonable assurance that the affected tube would not fail in the event of an accident. "Repair / plug in detection" is considered a subset of the depth based criterion. Additional basis information is provided in Draft Regulatory Guide 1.121.

In recent years, improved inspection techniques, knowledge of corrosion mechanisms, and experience have revealed additional types of tube degradation in the form of cracks in the tube wall. In some instances, a reliable method of characterizing specific types of cracks at defined locations within certain steam generator designs has been developed. In these cases, the industry has developed, and the NRC has approved Alternate Repair Criteria (ARC) to permit leaving a tube in service (as opposed to plugging) when the tube has indications that fall within the limits established by the ARC. Plug or repair on detection is not an ARC.

The NRC must approve all Repair Criteria prior to use. Requirements for approval of changes to the Repair Criteria are contained in the license Technical Specifications. New plugging designs or methods are not ARCs and do not require prior approval by the NRC. Repair Criteria approved for use at [Plant] are:

- [40%] nominal tube wall thickness
- [Other Repair Criteria that are currently approved for use – list.]

Steam generator tubes that exceed the Repair Criteria must be repaired or removed from service by plugging prior to entry into MODE 4 after the SG inspection that detected the degradation. This is necessary in order to provide reasonable assurance that tube integrity will be maintained until the next scheduled inspection. Due to technique and analyst uncertainties, sampling plans, and probability of detection there is a possibility that tube(s) exceeding the Repair Criteria will not be detected during a particular steam generator inspection. If the flaw(s) is detected during a subsequent inspection, the condition is not considered a reportable event unless it is determined that the Performance Criteria are not met.

BASES (continued)

Steam generator tube repairs are only performed using approved Repair Methods. 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 NRC must approve all Repair Methods prior to use. Requirements for approval of changes to the Repair Methods are contained in the license Technical Specifications. The Repair Methods approved for use at [Plant] are:

- [...]

This surveillance is required prior to entry into MODE 4 where the LCO is applicable. The surveillance requirement can be satisfied by determining that an acceptable Condition Monitoring / Operational Assessment exists and that the plant is within the operating cycle determined by the Operational Assessment.

REFERENCES

1. 10 CFR 50 Appendix A, GDC 19, Control Room
2. 10 CFR 100, Reactor Site Criteria
3. 10 CFR 50.72, Immediate notification requirements for operating nuclear power reactors
4. 10 CFR 50.73, Licensee event report system
5. NEI 97-06, Steam Generator Program Guidelines
6. EPRI Report TR-107621, *Steam Generator Integrity Assessment Guidelines*
7. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Rules for Construction of Nuclear Facility Components, Class 1 Components
8. Draft Regulatory Guide 1.121, Basis for Plugging Degraded

BASES (continued)

Steam Generator Tubes, August 1976

Enclosure 7

Template for a License Amendment Safety Analysis

Revision to NRC Submittal

Changes with respect to the February 4, 2000 version are bold, underlined, blue font.

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 eighteen 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.**

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 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 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
- 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]. 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] for the surveillance requirements necessary to verify tube integrity. In addition, the actions are changed to treat primary to secondary leakage the same as RCS pressure boundary leakage; shutdown must be commenced upon exceeding a leakage rate of 150 gallons per day in any steam generator. The previous proposed Technical Specification allowed 4 hours to reduce leakage to within limits. Finally, the surveillance requirements are revised to include notes that clarify mode applicability.

The changes to Administrative Technical Specification [5.5.9] require the implementation of a Steam Generator Program. 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 copy of the [Technical Requirements Manual] that includes the steam generator performance criteria defined in the Steam Generator Program and defines actions that would be necessary should the criteria not be met. The [Technical Requirements Manual] is provided for information only to illustrate how the Steam Generator Program requirements are incorporated into plant documentation. [Plant] is not requesting NRC approval of the [Technical Requirements Manual]. 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 Steam Generator Program required by the proposed license amendment. Note that many of the requirements discussed in the following section are part of the Steam Generator Program and are not specifically included in the Technical Specifications. The location of the requirement is provided in the third column of the table (TS = Technical Specifications, SGP = Steam Generator Program).

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Note
Operational primary to secondary leakage	[< 1 gpm total through all SGs and <500 gpd through any one SG]	TS - [≤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]	TS - Be in MODE 3 in 6 hours and in MODE 5 in 36 hours	2
Frequency of verification of tube integrity	6 to 40 months depending on SG category defined by previous inspection results	SGP - 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	SGP - Dependent on a pre-outage evaluation of actual degradation locations and mechanisms	4

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Note
Inspection techniques	Not specified	SGP - Performance based - dependent on a pre-outage evaluation of degradation types and NDE technique capabilities	5
Performance criteria	Operational leakage [<1 gpm total or < 500 gpd for any one SG]. No criteria specified for structural integrity or accident induced leakage.	TS – Operational leakage [≤150 gpd] through any one SG SGP - Structural integrity and accident induced 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	SGP - Requirements unchanged, but NRC approval is only required for first time use.	7
Failure to Meet Performance or Repair Criteria	Performance Criteria not defined. Operational leakage included in the Tech Specs. Plug or repair tubes exceeding repair criteria.	TS – Operational leakage included in the Tech Specs. SGP - 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	SGP - Unchanged except that NRC approval is required for first time use.	9

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Note
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.]	SGP - NRC reports required upon failure to meet a performance criterion or discovery of failure to implement required plugging or repair, and/or TS - 120 days after the initial entry into Mode 4 if >1% of the inspected tubes in any affected SG exceed repair criteria	10
Definitions	Normal Tech Spec definitions did not address SG Program issues.	SGP - Applicable SG Program definitions added to the [Technical Requirements Manual Bases].	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

If primary to secondary leakage exceeds [150 gpd]; plant shutdown must be commenced. Mode 3 must be achieved in 6 hours and mode 5 in 36 hours. There is no longer an allowance of 4 hours to reduce primary to secondary leakage to less than the limit.

The removal of the 4 hour period during which primary to secondary leakage can be reduced to avoid a plant shutdown is an improvement over the existing Operational Leakage Technical Specification. This change is consistent with the Steam Generator Program that also does not allow four (4) hours before commencing a plant shutdown.

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. The Steam Generator Program 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

The Steam Generator Program 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 NDE techniques that are effective in detecting their existence. Tube inspection techniques are chosen to reliably detect flaws that might progress during an operating cycle beyond the limits defined by the performance criteria.

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 Steam Generator Program. The existing Technical Specifications contain only the operational leakage criterion, therefore the proposed change is an improvement over the existing requirements. 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 Limiting Design Basis Accident. Any additional loading combinations shall be included as required by existing design and licensing basis.

The structural integrity 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 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 NRC has approved greater accident-induced leakage as part of a plant's licensing basis. Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria].

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 accident risk is sensitive to certain design basis parameters such as 1 gpm accident induced leakage. As a result, leakage greater than [Plant's] design basis or 1 gpm per steam generator is not allowed unless the NRC has approved greater leakage rates as part of an Alternate Repair 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 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 criterion 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] can be revised 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. Repair Criteria

The proposed license amendment requires 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 but is relocated to the [Technical Requirements Manual]. 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] can be revised to reflect approval of the criterion or method.

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

8. Exceeding Performance or Repair Criteria

The Steam Generator Program requires 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 [3.4.13]. Once shutdown, the 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 Steam Generator 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 during a previous SG inspection. 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 part of the [Technical Requirements Manual]. The existing Tech Specs addressed only operational leakage during operations.

During shutdown periods, the operational leakage criterion is no longer applicable, and the steam generators will be inspected as required by the licensee's program. A condition monitoring assessment of the "as found" condition of the steam generator tubes will be performed. 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 Steam Generator Program requires ascertaining the cause and determining corrective actions to prevent recurrence. Operation may resume when the requirements of the Steam Generator program have been met.

The proposed technical specification's change to the actions required upon exceeding the operational leakage criterion is conservative with respect to the existing Technical Specifications and is explained in Note 2 above.

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 Steam Generator Program requires 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 but is relocated to the [Technical Requirements Manual]. In addition, repair methods currently listed in the Technical Specifications are also relocated to the [Technical Requirements Manual]. However, the means of obtaining NRC approval of new repair methods is changed by this amendment.

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 will have the opportunity to approve all repair methods that are implemented by domestic PWRs. However, the proposed approach would eliminate the 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.

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 Bases] includes a number of new definitions that are important to the function of a Steam Generator Program. The new definitions and their explanations 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 cause the offsite radiological dose consequences required by 10 CFR Part 100 guidelines, or the radiological consequences to control room personnel required by GDC-19 to be exceeded.

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) Limiting Design Basis Accident, in the context of steam generator primary-to-secondary pressure boundary integrity, is the accident that results in either the largest differential pressure across the steam generator tubes for structural considerations or the minimum margin to the applicable dose limits for accident leakage considerations.

   The above definition does not change the intent of the limiting design basis accident; it just clarifies its application to the Steam Generator Program performance criteria.

4) 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 safety factor 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 safety factor of three against burst. Additionally, the 3 ΔP criterion is measurable through the condition monitoring process.

The actual operational parameters may differ between cycles. As a result of changes to these parameters, reaching the differential pressure in the equipment specification may not be possible during plant operations. Evaluating to the pressure in the design or equipment specification in these cases would be an unnecessary conservatism. Therefore, the definition allows adjustment of the 3 ΔP limit for changes in these parameters when necessary.

5) 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 a generically approved ARC at [Plant] will be justified by an evaluation that shows that

[Plant] design falls within the parameters defined by the NRC in the SER approving the ARC.

6) 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 a generically approved method at [Plant] can be justified by an evaluation that shows that [Plant] design falls within the parameters defined by the NRC in the SER approving the repair method.

7) Steam generator tubing refers to the entire length of the tube , including the tube wall and any repairs to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

   This definition ensures that all portions of 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 are an improvement to the existing SG inspection requirements and provide additional assurance that the plant licensing basis will be maintained between SG inspections. The proposed changes do 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.

Template for a License Amendment Significant Hazards Consideration

Revision to NRC Submittal

Changes with respect to the February 4, 2000 version are bold, underlined, blue font.

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. The proposed amendment requires a Steam Generator Program that defines a performance based approach to maintaining steam generator tube integrity. The Steam Generator Program includes Performance Criteria that define the basis for steam generator operability and provide reasonable assurance that the steam generator will remain capable of fulfilling its safety function of maintaining reactor coolant pressure boundary integrity. The existence of the Steam Generator Program is an improvement over the existing requirements.

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 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 Limiting Design Basis Accident. Any additional loading combinations shall be included as required by existing design and licensing basis.

The structural integrity performance criterion is a new requirement. It is documented in the Steam Generator Program.

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 location where the NRC has approved greater accident-induced leakage as part of a plant's licensing basis Exceptions to the1 gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria].

The accident induced leakage criterion is a new requirement. It is documented in the Steam Generator Program.

The operational leakage performance criterion is:

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

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

Probability of an Accident

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. The result of the implementation of the Steam Generator Program will be an enhancement of steam generator tube performance. 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 9

Draft Revision 1D to NEI 97-06

Changes with respect to Revision 0 are bold, underlined, blue font.

NEI 97-06 [Rev 1D]

Steam Generator Program Guidelines

DRAFT

Changes with respect to Revision 0 are bold, underlined, blue font.

November 2000

NEI 97-06 [Rev 1D]

Nuclear Energy Institute

Steam Generator Program Guidelines

November 2000

DRAFT

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.

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NOTICE

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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 to incorporate the improvements identified during the creation of the Steam Generator Generic License Change

Package and the resolution of technical issues raised by the NRC since the issuance of Revision 0. 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.**

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 **consider** 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.

Nuclear power plants licensed to operate prior to the effective date of 10 CFR 50, Appendix A (1971) are committed to the Proposed Appendix A to 10 CFR 50, General Design Criteria for Nuclear Power Plants, published in the Federal Register on July 11, 1967, which is similar to 10 CFR 50, Appendix A. Plant specific commitments to the Draft General Design Criteria are defined in the licensing bases for such plants.

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 Reactor Site Criteria

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

Primary-to-Secondary Leakage - Plant technical specifications include a requirement to shut down when primary-to-secondary leakage exceeds an established threshold.

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 requirements can be applied in operating plants. Meeting the intent of the EPRI Guidelines (References 2 through 7) is required. While it is recognized that specific site experience and steam generator design may require adaptation of select requirements within the supporting documents, the overall program elements are independent of steam generator design and apply to both first and second generation steam generators.

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 intent established by the applicable documents. Information on justifying deviations is provided in Appendix D.

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 this section except for the *PWR Steam Generator Tube Plug Assessment Document* [8]; and the *PWR Sleeving Assessment Document* [9].**

Additional information 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. Information on the NEI Steam Generator Review Board is provided in Reference 11.**

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. The NEI 97-06 document, or transmittal letter, will provide a listing of sections that have been revised along with the technical basis for the revision**.

When an EPRI Guideline is revised, EPRI SGMP will notify NEI who will notify PWR licensees. The SGMP transmittal letter, or the guideline document, will provide a listing of the revised sections of the guideline and the technical basis for each revision to the document's mandatory elements. Licensees will modify their steam generator programs during the time frame specified 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.

The steam generator performance criteria identify the standards against which performance is to be measuredPerformance criteria used for steam generators shall be based on tube structural integrity, accident-induced leakage, and operational leakage as defined below. Meeting the performance criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining RCPB integrity

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 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 limiting design basis accident. Any additional loading combinations shall be included as required by existing design and licensing basis .

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 stresses that the tube integrity assessments 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 generat<u>Leakage is not to exceed 1 gpm per steam generator, except for specific types</u> of <u>degradation at specific locationwhere the NRC has approved greater accident-induced leakage as part of a plant's licensing basis</u>. Exceptions to the 1-gpm limit can be applied if approved by the NRC in conjunction with approved alternate repair criteria.**

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

<u>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</u> not <u>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</u>.

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.

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 a plant's design basis or 1 gpm per steam generator is not allowed without prior NRC approval.

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.

The typical design basis accident analysis 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. The operational leakage performance criterion establishes a margin to the 1-gpm primary-to-secondary leakage value. Plant specific degradation mechanisms may exist which require a plant to implement reduced operational leakage limits

The PWR Primary-to-Secondary Leak Guidelines [3] provide reasonable assurance that the operational leakage performance criterion will be met. Measurement and detection methods and associated actions shall adhere to the intent of these guidelines.

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 is maintained. Observance of these elements is required to implement NEI 97-06 as intended.

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 so 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 take actions in accordance with plant procedures, including notifying the NRC as applicable**.

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**. The 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 alternate repair criteria (ARC). If licensees choose to develop and implement an ARC, **they should follow a steam generator degradation-specific management (SGDSM) strategy. NRC approval is required prior to a plant's initial use of an ARC**.

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 and is not considered an ARC**.

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 prior to implementation. New plugging designs or methods do not require prior approval by the NRC.

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 **PWR** *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

In addition to the utility-specific technical specification reporting requirements, the following reports are required if the information has not already been reported:

Condition	Reports Required
Results of the steam generator inspection indicate greater than 1% of the inspected tubes in any steam generatorexceed the repair criteria.	**Submit Condition Monitoring including results of tube pulls and in situ testing within 120 days after the RCS reenters Hot Shutdown conditions.**
Failure to meet a performance criterion discovered during condition monitoring.	**Submit an Operational Assessment establishing the basis for the next operating cycle.** **Assess to determine if a degradation of a safety barrier has occurred, if so, it is considered a reportable event - notify the NRC in accordance with the requirements of 50.72 and 50.73**

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 thdong term viability of the steam generator. Note that observance of these elements is required to adhere to the intent of 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 Reporting includes internal reports that document information within the plant's Steam Generator Program and external reports intended to be shared with other utilities.

Internal reports include Degradation Assessments, Tube Integrity Assessments, and technical justifications for deviation from SG Program requirements. These reports should be retained as records within the SG Program.

External 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.

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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 A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity* (September 1988).**

11. ***Steam Generator Management Program Administrative Procedures*, Product ID 1000776 *.**

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.

Alternative RepairCriteria(ARC)

Alternative Repair Criteria (ARC) are tube repair criteria that may be implemented for a specific defect type as part of a Steam Generator Degradation Specific Management (SGDSM) program in lieu of the generally applicable depth-based criterion. (Plug on detection is not an ARC).

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

In the context of steam generator primary-to-secondary pressure boundary integrity, it is the accident that results in either the largest differential pressure across the steam generator tubes for structural considerations or the minimum margin to the applicable dose limits 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

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 flow induced forces should be considered.

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, including the tube wall and any repairs to it, between the tube-to-tube sheet weld at the tube inlet and the tube-to-tube sheet weld at the tube outlet. The tube-to-tube sheet 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

DRAFT

Appendix D

TECHNICAL JUSTIFICATION REQUIREMENTS FOR DEVIATION FROM NEI 97-06 REFERENCED DOCUMENTS

Introduction

NEI 97-06 requires that a utility meet the intent of the referenced EPRI guidelines. These documents contain industry "best practices" and provide guidelines on procedures, testing, and methodology, the intent of which if followed will help assure compliance with the measures. In some cases, these documents delineate specific elements that may consist of control variables or action levels, procedures, processes, or testing that must be explicitly met by the utility to meet the intent of NEI 97-06. The guidelines and information documents are listed at the end of this appendix.

The purpose of this appendix is to identify levels of documentation that must be developed and reside in a steam generator program if deviations are taken from any of the mandatory elements, procedures, testing, methodology, or guidelines presented in the EPRI guidelines referenced in NEI 97-06. As noted above, some of the support documents (e.g., PWR Primary Water Chemistry Guidelines and the PWR Steam Generator Examination Guidelines) contain elements that must be implemented by the utility. These elements are called "mandatory" in this appendix and are discussed in Section 3 of Reference 11. Deviation from these elements require a technical justification in accordance with this appendix. Additionally, it is suggested that deviations from non-mandatory procedures, testing, methodology, or guidelines listed in the guidelines referenced in NEI 97-06 be justified and documented in the utility's steam generator program.

General Considerations

1. When it is determined that a proposed procedure, testing, methodology or guideline will be implemented in a plant's steam generator program which may not be consistent with the mandatory elements documented in the applicable NEI 97-06 referenced guideline, technical justification shall be written and made part of the utility's steam generator program. Note, a utility's steam generator program includes all sub-tier programs such as the chemistry strategic and optimization programs. Alternatively, the justification may be included in one of these sub-tier programs to which it is applicable. Additionally, deviations from the intent of any other procedure, testing, methodology or guideline presented in NEI 97-06 referenced guidelines requires a technical justification which shall be

written and approved in accordance with the utility's steam generator program and made part of that program. Deviation from specific elements of NEI 97-06 referenced guidelines, such as procedures, testing, methodology or guidelines, which is not inconsistent with the original intent of these elements, should be documented with justification in the utility's steam generator program.

2. If the devia tion deals with an integrity element specified in NEI 97-06, the technical justification shall provide the basis for the determination that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the performance criteria of NEI 97-06.

3. In no case shall any deviation allow a change to a plant's licensing basis without review and approval required by regulation.

4. To help assure that a thorough review of a deviation and its' justification associated with any procedure, testing, methodology, or guideline documented in NEI 97-06 referenced guidelines is performed, it may be necessary to solicit comments or input from a number of sources. These sources should not always be limited to in-house utility personnel, but other utility/industry experts should be consulted where appropriate. An appropriate situation for outside review may be that associated with inspection of a degradation mechanism that is first identified in the plant's steam generator and for which a deviation on inspections processes is developed. Additionally, deviations from mandatory elements or intent of the guidelines shall be given this type of review.

5. If the suggested deviation involves an interpretation of any requirement procedure, testing, methodology, or guideline documented in NEI 97-06 referenced guidelines, the NEI Review Board addressing interpretations of NEI 97-06 shall be consulted for assuring the accuracy of the interpretation. Such interpretations should be addressed in this manner and resolved before the deviation involving an interpretation is implemented by approval authority in the steam generator program.

Technical Justification for Deviation Document Structure

1. The subject justification should contain a description and location of the affected requirement.

2. The justification should be explicit in detailing the deviation that is being taken. It should clearly identify what applicable additional information, potential and acceptable impact to the plant, or knowledge is available that allows the deviation to be acceptable. If the deviation deals with an

integrity element specified in NEI 97-06, the justification shall demonstrate the deviation maintains the same level of performance criteria required under NEI 97-06.

3. Th e accompanying justification for the deviation should clearly state how long this deviation will be in force.

4. The justification should clearly identify all applicable site organizations that the deviation may impact in terms of procedures or decision making processes, such as NDE, water chemistry, and operator personnel.

5. The justification or document in which it is contained should clearly identify appropriate approval signatures and dates per the utility's document control procedures.

EPRI guidelines referenced in NEI 97-06 are:

1. 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. Steam Generator In-Situ Pressure Test Guidelines

Information documents referenced in NEI 97-06 are:

1. PWR Steam Generator Tube Plug Assessment Document

2. EPRI PWR Sleeving Assessment Document