Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines

Revision 3

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Non-Proprietary Version

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REPORT SUMMARY

This report provides guidance for evaluating the condition of steam generator (SG) tubes based on nondestructive examination (NDE) or *in situ* pressure testing. The integrity assessments are normally performed during a reactor refueling outage. Nuclear power plant licensees who follow the guidance in this report will have satisfied the requirements for degradation assessments, condition monitoring, and operational assessment as defined in the Nuclear Energy Institute (NEI) *Steam Generator Program Guidelines*, NEI 97-06.

Results and Findings
This revision incorporates operating experience since Revision 2 of this guideline. The following major changes have been incorporated:

- Important terms were defined and made consistent with other guidelines.
- Guidance from Steam Generator Management Program (SGMP) Interim Guidance SGMP-IG-07-01 was added to appropriate sections.
- Guidance from SGMP Information Letters SGMP-IL-06-01 and SGMP-IL-07-01 were added to appropriate sections.
- The technical basis for default growth rates was added in Section 5.
- Recent cracking experience in thermally treated Alloy 600 tubing was added.
- Full bundle and worst-case analyses were clarified.
- Equations were clarified and independent review of equations and examples was provided.

Challenges and Objectives
- To help licensees support the implementation of NEI 97-06
- To help licensees evaluate the condition of the SG relative to the performance criteria in NEI 97-06 and the plant's SG Technical Specifications
- To define requirements and describe in detail the implementation procedures for successful SG integrity assessments

Applications, Value, and Use
This report describes acceptable methods for degradation assessments, condition monitoring, operational assessments, and secondary-side assessments. There are other acceptable methods for integrity assessment; however, technically justifying their application would be the utilities' responsibility. Where alternative repair criteria are approved for use by the U.S. Nuclear

Regulatory Commission (NRC), specific assessments are required; therefore, alternative repair criteria are not covered in this guideline. Utilities' SG program owners should use the information in this guideline to satisfy the requirements for tube integrity assessments in NEI 97-06 and in Technical Specifications.

EPRI Perspective

NEI 97-06 requires condition monitoring and operational assessment of SG tubing. This report provides a useful description of the way in which SG tubing can be shown to meet required performance criteria. Using a standard approach facilitates acceptance and review by regulatory authorities.

This report reflects current industry practices and represents an acceptable method for performing integrity assessments. Revisions can be expected as the industry accumulates experience with this guideline.

Approach

This revision was developed using a committee of industry experts.

Keywords

Nuclear steam generators
Degradation assessment
Condition monitoring
Operational assessment
Secondary-side integrity assessment
Integrity assessment
Leakage assessment

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1 INTRODUCTION

1.1 Objective

These guidelines present requirements and implementation procedures for meeting the objectives of steam generator tube integrity assessments including:

1. Identification and characterization of degradation forms within steam generators that require assessment,
2. Application of appropriate NDE technology, consistent with the existing and potential degradation and in accordance with the SGMP PWR Steam Generator Examination Guidelines [1].
3. Application of integrity assessment methods, consistent with the existing and potential degradation and required safety factor, for use in evaluating integrity at the end of an inspection interval and ensuring integrity during the subsequent inspection interval.

Successful implementation of the above objectives will help ensure that steam generator integrity will be maintained for each degradation form during the full range of normal operating conditions and applicable design basis accidents.

Licensees should use this document to demonstrate the condition of their steam generators relative to performance criteria used for condition monitoring (CM) and operational assessment (OA) as defined in NEI Steam Generator Program Guidelines, NEI 97-06 [2].

The U.S. nuclear power industry established a framework for increasing the reliability of steam generators by adopting NEI-97-06 [2]. This initiative references the Steam Generator Integrity Assessment Guidelines as the basis for an industry consensus approach to demonstrating the condition of steam generators relative to the performance criteria. Specifically, the initiative requires that US nuclear power plant licensees meet the intent of the Steam Generator Integrity Assessment Guidelines. The focus of the NEI initiative is steam generator integrity. All US plants have adopted Steam Generator Technical Specifications based on NEI 97-06 (TSTF 449 [45]).

The Steam Generator Management Program (SGMP) Administrative Procedures [3] include protocol that is to be followed by guideline revision committees with regard to establishing the level of implementation expected by a licensee. In particular, three categories or elements have been established: mandatory requirements, shall requirements, and recommendations. These categories are clearly defined in reference 3. These categories are summarized below, using quotations from reference 3 when feasible.

1. "Mandatory requirements ... are important to steam generator tube integrity and should not be deviated from by any utility. Steam generator tube integrity is described as meeting the performance criteria as specified in NEI 97-06" (discussed in Section 2 of these guidelines). There are two mandatory requirements within the document and they are listed and highlighted in bold print in Section 12.
2. "Shall requirements ... are important to long-term steam generator reliability but could be subject to legitimate deviations due to plant differences and special situations." Shall requirements appear throughout the document and are listed in Section 12.
3. The SGMP Administrative Procedures [3] provide the process for addressing deviations to mandatory and shall requirements.
4. "Recommendations are good or best practices that licensees should try to implement when practical. Written documentation is not required when a recommendation is not implemented." Recommendations are listed in Section 12.

If there is a conflict between the steam generator program, as provided in NEI 97-06 and supporting documents, and the Plant Technical Specifications, the latter shall govern.

The performance criteria address structural tube integrity, postulated accident leakage, and operational leakage, and are discussed in Section 2 of this document. Internal and external reporting requirements are detailed in Section 11.

1.2 Scope

As required by NEI 97-06, this document offers guidance and requirements for the evaluation methods, margin, and uncertainty considerations used to determine tube integrity. It also provides guidance for performing steam generator degradation assessments (DA), condition monitoring (CM), operational assessments (OA), and secondary side assessments. Information on how to carry out these assessments is provided in the body of this document with supplemental examples in Appendix E. These methods are proven and supported by industry experts. Other approaches may be used but must be technically justified.

1.3 Basic Methodology of Steam Generator Integrity Assessment

This section summarizes the details of steam generator integrity assessment. This assessment applies to steam generator components which are part of the primary pressure boundary (e.g., tubing, tube plugs, sleeves and other repairs). It also applies to foreign objects and secondary side structural supports (e.g., tube support plates) that may, if severely degraded, compromise pressure-retaining components of the steam generator.

The essential elements of steam generator integrity assessment are presented in Figure 1-1. Three critical elements, the DA, CM, and OA, provide assurance that the steam generators will continue to satisfy the appropriate performance criteria.

The degradation assessment is the planning process that identifies and documents information about plant-specific steam generator degradation. The overall purpose of the DA is to prepare for an upcoming steam generator inspection through the identification of the appropriate examinations and techniques, and ensuring that the requisite information for integrity assessment is obtained. Discussion of the DA is provided in Section 6.

Condition monitoring is backward looking, in that its purpose is to confirm that adequate steam generator tube integrity has been maintained during the previous inspection interval. Condition monitoring involves an evaluation of the as-found condition of the tubing relative to integrity performance criteria. The tubes are inspected according to the EPRI PWR SG Examination Guidelines [1]. Structural and leakage integrity assessments are performed and results compared to their respective performance criteria. If satisfactory results are not achieved, a root cause evaluation is performed and appropriate corrective action taken. The results of this analysis are factored into future DAs, inspection plans, and OAs of the plant. Condition monitoring strategies are provided in Section 7.

The operational assessment differs from the CM assessment in that it is forward looking rather than backward looking. Its purpose is to demonstrate that the tube integrity performance criteria will be met throughout the next inspection interval. Integrity performance criteria are provided in Section 2 and OA strategies are provided in Section 8.

Integrity assessment of the steam generator secondary side is necessary to verify that tube safety functions are not jeopardized by foreign material or internals degradation (Section 10).

Figure 1-1 shows how steam generator integrity can be monitored and maintained. During Condition Monitoring, inspection results are to be evaluated with respect to the appropriate performance criteria. Condition monitoring results are then to be evaluated with respect to the previous OA. If this evaluation is successful, an Operational Assessment is made to show that integrity will be maintained throughout the next interval between inspections. If any performance criterion is not met during performance of Condition Monitoring, then the licensee shall perform a root cause evaluation, submit reports to the NRC according to plant reporting requirements, and factor the results into the OA strategy. The results of OA determine the allowable run time for the upcoming inspection interval.

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Figure 1-1
Steam Generator RCS Pressure Boundary Assessment; Degradation Assessment, Condition Monitoring and Operational Assessment

1.4 Compliance Responsibilities

This document presents a general approach and examples for demonstrating steam generator tube integrity. Plant specific programs should consider plant design, materials, steam generator corrosion experience, and operating philosophy. Performing the assessments herein will help plant personnel understand what inspections and repairs are necessary and the appropriate length of operation between inspections. To meet this goal, an effective corporate policy and monitoring program are essential and should be based on the following:

- Clear management support for plant procedures including the requirements and guidance in this document.
- Adequate staff and organizational resources to implement an effective steam generator program.
- Continuing review of plant and industry experience and research results to revise the program as warranted.

1.5 Contractor Oversight

While work can be contracted to carry out the requirements of this document, the responsibility for implementation and compliance resides with the licensee. It is the licensee's responsibility to plan, direct, and evaluate SG examination activities. The licensee oversees not only the contractual, but also the technical aspects of any contracted work in accordance with NEI 97-06.

2
TUBE INTEGRITY CRITERIA

2.1 Introduction

This Section presents analysis margins and acceptance criteria for structural integrity and through-wall leakage associated with degraded steam generator tubing. The conditions for demonstrating tube structural integrity are defined by the structural integrity performance criterion (SIPC). The SIPC defines the margins to be applied in maintaining adequate tube integrity against gross failure by either burst or plastic collapse. The technical bases for the SIPC follow from the design margins implied in Section III of the ASME Boiler and Pressure Vessel Code. Leakage integrity performance criteria are defined separately for operational and accident-induced conditions. It is mandatory that CM and OA be performed to demonstrate overall tube integrity. Tube integrity is demonstrated by satisfying the structural integrity and leakage performance criteria in conjunction with the performance acceptance standards described in Section 2.4.

2.2 Structural Integrity Performance Criterion

The SIPC provides the margins for tube integrity against tube burst or collapse. The structural integrity performance criterion is:

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Figure 2-1
SIPC Implementation Logic

2.3 Leakage Integrity Performance Criteria

The leakage integrity performance criteria provide requirements for both operational and accident leakage. The operational leakage performance criterion is:

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The accident-induced leakage performance criterion (AILPC) is:

2.4 Performance Acceptance Standards

The performance acceptance standards for assessing tube integrity to the structural integrity and accident leakage performance criteria apply to both CM and OA. The acceptance standard for structural integrity is:

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2.5 Discussion of Structural Margins and Bases

2.5.1 Assessment Factors

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2.5.2 Burst Definition

Steam generator tubes must exhibit a low probability of burst under normal operating conditions and accident conditions. The definition of tube burst is:

This definition is discussed in Appendix C.

2.5.3 Collapse Definition

Steam generator tubes must also exhibit a low probability of failure by net-section collapse under accident conditions. Tube collapse refers to the failure condition where a tube forms a plastic hinge under the combined action of axial and bending loads. For purposes of this document, the definition of tube collapse is defined as:

2.5.4 Limits on Yield Strength

Historical regulatory precedence dictates that tube integrity be maintained for the full range of reactor operation including startup, operation in the power range, hot standby and cool down, and all anticipated transients that are included in the design specification. Hence, a key aspect of tube integrity as stated in NEI 97-06 [2], is to ensure that degraded steam generator tubes are not stressed beyond the elastic range of the tube material for primary membrane loads for normal operating conditions (i.e., gross yielding of the tube is prevented during normal operation). In this context, tube integrity for all expected operating scenarios (i.e., the full range of reactor operation including startup, operation in the power range, hot standby and cool down, and all anticipated transients) is demonstrated if the associated primary membrane stresses are below the yield strength of the tube material for Service Levels A and B (normal and upset) conditions.

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2.6 Pressure Load Definitions

The important loading condition for structural integrity performance assessments is the pressure differential across the tube wall during normal steady state operation and accident conditions as defined in the SIPC. For normal plant operation, the pressure across the steam generator tube is the primary-to-secondary pressure differential occurring at normal full power operation.

Normal steady-state full power operation as defined in NEI 97-06 [2], and discussed in Appendix C, is:

The limiting accident pressure differential is the maximum or largest pressure differential across the tube wall for the design basis accidents (Service Levels C and D). For most plants, this is the pressure differential during a main steam line break. Apart from pressure loading, other contributing loads that can occur during the postulated accidents shall be evaluated to determine if these loads contribute significantly to tube burst. Such loads are discussed in Section 2.5.1, Section 3.7.2, and in Appendix C.

3 TUBE INTEGRITY ASSESSMENT LIMITS

3.1 Introduction

This Section presents the general requirements for establishing the tube integrity and repair limits associated with steam generator tubing, such that the tube integrity performance criteria and the performance acceptance standards defined in Section 2 are satisfied during operation. Tube integrity limits are defined for each existing and potential degradation mechanism.

Determination of tube integrity limits includes identifying an acceptable structural parameter, such as loss of wall thickness or degradation length, which can be related to tube integrity and can be measured by NDE technology. This process reviews the practicality of the selected NDE technique for the chosen structural parameter, including estimation of the probability of detection (POD) of a flaw and the uncertainties in quantification of flaw sizes. The determination of these limits can be performed either deterministically or probabilistically as discussed later in Sections 7 and 8.

3.2 Tube Integrity Limits

The focus of this Section is on the determination of structural limits to prevent tube burst. Leakage integrity limits are discussed in Section 9. A structural limit is established from the mean (best estimate) regression relationship for tube failure for the conditions defined by the SIPC. The condition monitoring limit is obtained by modifying the structural limit to account for the uncertainties associated with the tube failure regression model, material properties, and the NDE system. The repair limit and OA limit are obtained by further modification to consider degradation growth, and require that flaws on tubes remaining in service at the beginning of cycle (BOC) satisfy the structural integrity performance criterion over the next inspection interval.

The SG Degradation Specific Management Flaw Handbook [5] provides a compilation of algorithms for determining plant-specific structural limits for degradation mechanisms for which structural limits have been defined. The structural parameter may be defined explicitly in terms of a mechanical variable, such as crack length and/or depth, or by percent degraded area. Alternatively, this structural parameter may be defined in terms of an implicit variable that is empirically related to the strength of the tube (e.g., eddy current bobbin voltage, RPC voltage, etc.).

USNRC Draft Regulatory Guide 1.121 [46] provides another methodology to establish structural limits.

3.2.1 Condition Monitoring Limit

Condition monitoring is the assessment of the current state of the steam generator tubing, and is performed at the conclusion of each steam generator inspection. The purpose of the CM is to confirm that both the structural integrity and accident-induced leakage performance criteria were satisfied during the past inspection interval. The assessment involves a comparison of the as-found inspection results against the performance acceptance standards for structural integrity and accident leakage. Because the detected indications, in terms of the distribution of either indication voltages or measured flaw sizes, reflect a conservative estimate of the in-service population of flaws at the end of the cycle (EOC), monitoring the as-found condition will provide a conservative evaluation of the current condition of the tube bundle, including the flaws that remain undetected after inspection. In this situation, the performance acceptance standard can be applied to the detected population to verify the steam generators met the SIPC during the previous inspection interval. Analysis methods to perform this evaluation are discussed in Section 7.

Condition monitoring therefore requires that the detected flaws, as determined by in-service inspection, do not exceed the appropriate CM limit for each degradation mechanism. A schematic illustration of the CM structural limit is shown in Figure 3-1. Its determination involves calculations with the following parameters:

1. A burst model based on regression analysis of tube failure data, including uncertainty in the prediction of burst pressure, for a given extent of degradation,
2. Tube material strength information, including uncertainty in mechanical strength behavior due to material heat-to-heat and within-heat variability, and
3. Measurement uncertainty to the NDE sizing technique, including systematic and random components of sizing uncertainty due to technique and analyst variability.

Condition monitoring calculations can be performed following the methods described in Section 7.

3.2.2 Operational Assessment Limit

An operational assessment is a forward-looking prediction of the steam generator tube conditions at the next inspection. Operational assessments require that the projected sizes of the undetected population of flaws, or detected flaws intentionally left in service as determined by analysis, do not exceed the OA limit. The operational assessment limit is the value of the degradation parameter such that a tube with greater degradation would not meet the SIPC at the next SG inspection.

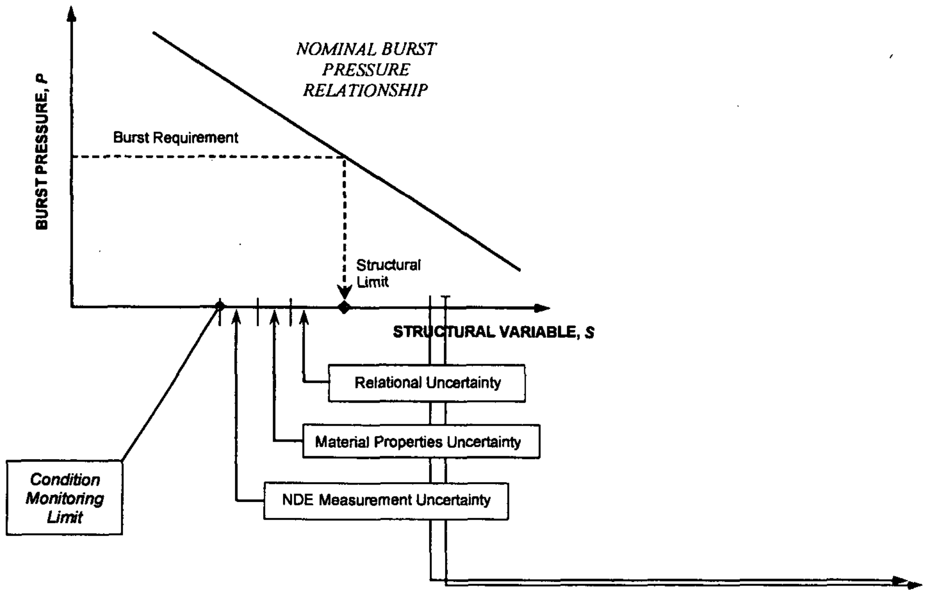

Figure 3-1
Condition Monitoring Elements of Tube Integrity Assessment

Operational assessment requires that the degree of degradation of any flaw remaining in service does not exceed the appropriate OA limit for any degradation mechanism. A schematic illustration of an OA is shown in Figure 3-2 for repair on NDE sizing and Figure 3-3 for repair on detection. The repair limit and OA limit are similar in nature in that they provide the beginning of cycle (BOC) structural parameter to meet the SIPC at end of cycle (EOC), depending on whether one is using sizing or probability of detection (POD) to establish the BOC requirement.

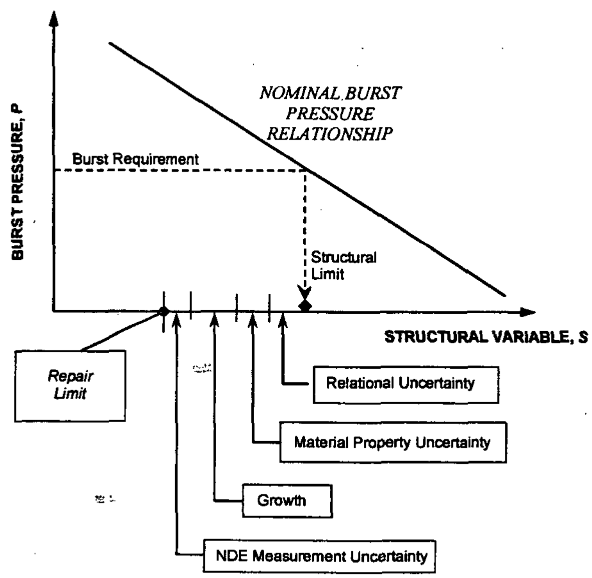

Figure 3-2
Operational Assessment Elements of Tube Integrity (Repair on Sizing)

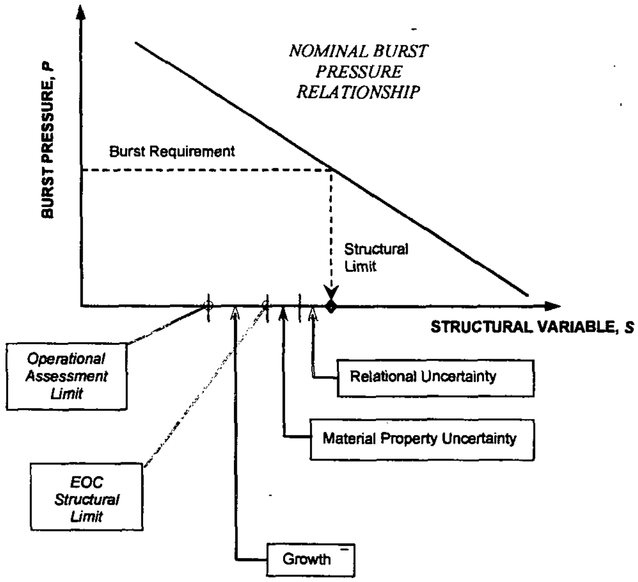

Figure 3-3
Operational Assessment Elements of Tube Integrity (Repair on Detection)

Determination of the repair and OA limits involves calculations with the following:

1. A burst model based on regression analysis of tube failure data including uncertainty in the prediction of burst pressure for a given extent of degradation,
2. Tube material strength information including uncertainty in mechanical strength behavior due to material heat-to-heat variability (to establish the EOC Structural Limit shown in Figure 3-3) as defined in Reference [5],
3. Degradation growth during future operation, and
4. For repair on NDE sizing, measurement uncertainty associated with the sizing technique, including systematic and random components of sizing uncertainty due to technique and analyst variability.

It should be noted that the repair and operational assessment limits are sometimes referred to as the BOC Allowable Limit.

The operational assessment limit can be determined by the methods described in Section 8. These methods include deterministic and probabilistic techniques. Also discussed in Section 8 are the conditions for which a deterministic analysis may not be adequate to perform a conservative OA. When a deterministic approach is used following the methods and input assumptions defined in Section 8, the OA limit can be determined on a limiting flaw basis, where a worst case flaw is projected to exist at the time of the next inspection.

3.3 Material Properties

The structural limit depends upon the flow strength of the material, which is a function of both the yield and ultimate tensile strengths for the tubing. The integrity assessment limits shall include adjustments to account for statistical distributions associated with tube material properties. Both CM and OA limits reflect the uncertainty in strength determined at the operating temperature for the as-fabricated generator. The Flaw Handbook [5] provides tables that give statistical distribution information at room temperature and elevated temperatures for various tubing sizes, material specifications, and suppliers. Alternatively, actual material properties may be used. ASME Code minima may also be used if material property distributions for the actual tubing are not known. If the ASME Code equation is used, Code minimum properties from the Code of Record for the plant's licensing basis shall be used. Material properties should be determined at 650°F, or as justified for plant-specific analysis assumptions.

3.4 Repair Limit

The repair limit is the NDE measured parameter at or beyond which the tube is to be repaired or removed from service by plugging. It is recommended that repair limits for existing degradation mechanisms be established prior to the inspection and tubes exceeding this limit during the inspection shall be repaired. The repair limit is defined such that the performance criteria will be met at the end of the inspection interval. If a Technical Specification repair limit is defined, the more limiting value shall be used.

3.5 Technical Specification Repair Limit

A plant's Technical Specification repair limit (e.g., 40% through-wall) is determined from bounding values of NDE sizing uncertainty and degradation growth rate in conjunction with a conservative safety margin. Flaws that are smaller than the repair limit may be left in service when technically justified by the OA. Historically, the 40% repair limit has been derived by subtracting allowances for NDE uncertainty and degradation growth from a structural limit. The repair limit is based on maximum measured depth. Each plant shall verify that its Technical Specifications depth-based repair limit is applicable to the type of tube degradation to which it is being applied, and that the assumptions used to derive the repair limit are justified.

3.6 Special Considerations for Tube Integrity Assessment

There are situations where physical constraints can be used to presume compliance with the tube integrity performance criteria. One example of such a physically based argument would be the case of a tube with its degraded section confined within a tube support plate that can be shown not to move during the limiting design basis accident. It can then safely be assumed that the tube will not burst in this situation, although it might leak. In this case, structural integrity is met by definition, but leakage integrity still needs to be evaluated. Similar conditions could be established that would limit both tube burst and leakage, such as degradation detected within the tube sheet. Whenever credit is taken for physical constraints, the conditions shall be technically justified.

tube sheet. Whenever credit is taken for physical constraints, the conditions shall be technically justified.

3.7 Determination of Structural Integrity Limits

3.7.1 Tube Burst Event

In applying the SIPC, structural limits for allowable tube wall degradation are determined for three sets of postulated conditions, as illustrated in Figure 2-1. The conditions for determining allowable burst pressure, or allowable structural limits on degradation, are defined by separate analyses. The limits for accident conditions are determined from a tube failure model that considers pressure differentials, and combined primary loads (bending plus pressure) and axial secondary loads and their synergistic effect on burst pressure and/or plastic collapse.

The structural limit for a given degradation mechanism is the maximum degradation size that satisfies all three requirements. In most situations, the structural limits are set by the safety factor of three imposed on full power steady state operating pressure. In some situations, the establishment of the structural limit will be governed by accident conditions. This will be dependent on plant design, the specific degradation mechanism and location, and the applicable set of transient parameters for the plant under consideration. In a few situations, contributing non-pressure loads may be significant.

3.7.2 Significant Contributing Loads

Potential contributing non-pressure primary loads created during accident conditions would be those bending loads resulting from dynamic conditions. Dynamic conditions include cross flow and other hydraulic and inertia forces from LOCA and seismic events. Both recirculating steam generators (RSGs) and once-through steam generators (OTSGs) are subjected to these types of primary bending loads.

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Figure 3-4
Logic for Screening Contributing Loads

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3.7.3 Tube Collapse Event

Tube collapse by net section plastic failure under combined tension and bending loads was also evaluated in Reference [6].

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4
NDE MEASUREMENT UNCERTAINTIES

4.1 Introduction

This Section addresses eddy-current NDE system performance measures and their uncertainties associated with steam generator tube bundle examination for tube integrity applications. Two aspects of performance are considered: 1) degradation detection, quantified by Probability of Detection (POD), and 2) degradation sizing, quantified by linear correlations of true-versus-measured values of structural quantities of interest, such as length and depth of degradation.

Total system NDE performance uncertainties include technique and analyst variability. For all existing and potential degradation mechanisms identified in the DA, total NDE system performance shall be established for detection and sizing.

Table 4-1 lists the various tube wall degradation mechanisms that have been confirmed in domestic NSSS steam generators. Two broad categories are recognized; 1) corrosion related and 2) mechanically induced. Corrosion degradation includes thinning, pitting, outer diameter stress corrosion cracking (ODSCC) and primary water stress corrosion cracking (PWSCC) while mechanical degradation includes wear, impingement and high-cycle fatigue.

For older plants with original SGs, the focus is generally on cracking mechanisms, i.e., ODSCC and PWSCC. Repair on detection is done for such degradation, unless plant-specific alternate repair criteria (ARC) have been licensed. Wear is usually the focus for plants with newer or replacement steam generators. Repair on sizing is generally done for wear.

Thinning and pitting are of historical interest for Combustion Engineering design units but are no longer deemed relevant. The same is true for pitting in Westinghouse design units and impingement in Babcock & Wilcox design units. High-cycle fatigue cracking is not normally addressed during eddy-current tube bundle examination because of excessive growth rates.

Section 4.2 defines POD requirements and limitations on POD for tube integrity applications, and discusses POD modeling. Examples are provided to show how the requirements can be met, with additional examples provided in Appendix D. Section 4.3 discusses sizing requirements and limitations. Extensions of examination techniques are discussed in Section 4.4.

Table 4-1
Steam Generator Tube Wall Degradation

Degradation Mode	W	CE	B&W	Repair on Sizing?
Corrosion				
Thinning	x	X[1]	N/A	Yes
Pitting	X[1]	X[1]	N/A	NA
ODSCC	x	x	x	No[2]
PWSCC	x	x	x	No[2]
Mechanical				
Wear	x	x	x	Yes
Impingement	N/A	N/A	X[1]	NA
High-Cycle Fatigue	x	x	x	NA

[1] Historical; no longer active or relevant

[2] Unless alternate repair criteria are licensed.

4.2 Probability of Detection

4.2.1 Requirements and Limitations for Tube Integrity Applications

An NDE system is defined as the equipment, procedure and personnel used in performing an inspection. Steam generator tube examination technique qualification, as currently specified in Appendix H of the SGMP PWR Steam Generator Examination Guidelines [1] and associated Examination Technique Specification Sheets (ETSS) [28], focus on a portion of the system i.e., equipment and procedure. Personnel effects, such as performance variations expected with multiple data analysts (human factors), and the impact of blind testing, have historically not been addressed. Accordingly, existing ETSS data sheets provide technique rather than system performance estimates for detection. The EPRI Tools for Integrity Assessment Project [7-8] approach for determination of the entire NDE system performance indices is now addressed in Appendix I of the SGMP PWR Steam Generator Examination Guidelines [1]. There will be a transition period where both Appendix H and Appendix I ETSSs are available.

The Tools for Integrity Project has developed a system POD for one of the degradation mechanisms (ODSCC) using a Performance Demonstration. This process will continue until all ETSSs include system performance uncertainties. These ETSSs or site specific ETSSs developed to the rigor of Appendix I shall be used if applicable for the assessment of tube integrity as they become available.

The PODs were established using the empirical approach described in Section 4.2.4. The Tools for Integrity Assessment Project team has evaluated a number of proposed POD functions, and recommends the following non-linear log-logistic function:

$$POD(S) = \frac{1}{1 + e^{-(a + b \log S)}}$$ **Equation 4-1**

where S is the structural parameter (e.g., length, depth, or percent degraded area), and a and b are model parameters to be determined. In general, these parameters are degradation mechanism and examination technique specific. This POD function is discussed in Sections 4.2.2 and 4.2.3.

The Tools for Integrity Assessment Project also evaluated the effects of noise on POD. The empirical PODs and the log-logistic function described above are noise dependent, and plant-specific noise amplitude distributions, which may in turn be location-specific, should be used to determine the impact of noise on POD. A method for doing so is discussed in Appendix D, Section D.1.2.6, and Reference [9].

4.2.2 POD Modeling

A POD model is a functional measure of the ability of an NDE system to detect degradation. It is one of the inputs to an OA and is used to estimate degradation remaining in service after an eddy-current examination of the tube bundle. As shown in Figure 4-1, a POD model is constructed using binary hit-miss data in which NDE detections and non-detections, coded as 1's and 0's respectively, are plotted using a structural parameter as the independent variable, and then non-linearly regressed using an appropriate mathematical function which is defined as the POD model. It should be noted that the resulting POD model is dependent on the available hit-miss data and its distribution with respect to the structural variable. Various structural parameters such as maximum depth, average depth, length, voltage, etc. can be used as the independent variable depending on the needs of the tube integrity engineer.

Binary hit-miss data can be regressed using different non-linear functions resulting in different POD models for the same input. This point is illustrated in Figure 4-2 where the same hit-miss data have been fit using logistic and log-logistic functions. Note that for this data set, the logistic function is non-zero at the origin with a value of ~1 at the upper extreme of the independent variable whereas the log-logistic function is fixed at the origin but assumes a value <1 at the upper extreme. Traditionally, logistic functions were generally used in the early days of OAs for POD modeling; currently the use of log-logistic functions is more prevalent because of the benefit of fixing the function at the origin. The lower POD value at the upper extreme of the structural variable is viewed as being conservative for OA applications, as compared with what is obtained using a logistic function. Since a log-logistic model is currently the most common model in use, its use is recommended for standardization purposes.

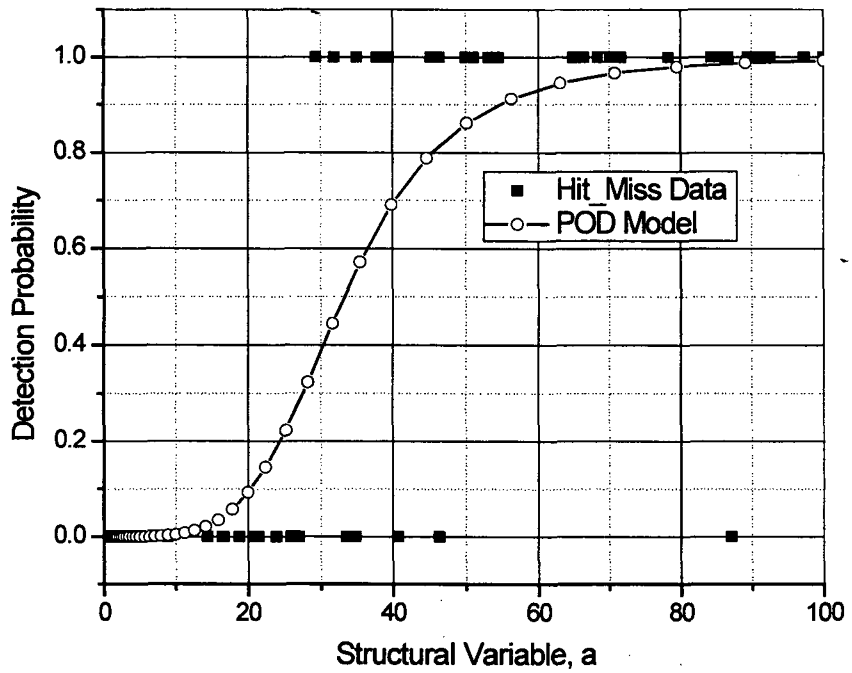

Figure 4-1
Generating a POD Model Using Binary Hit-Miss Data

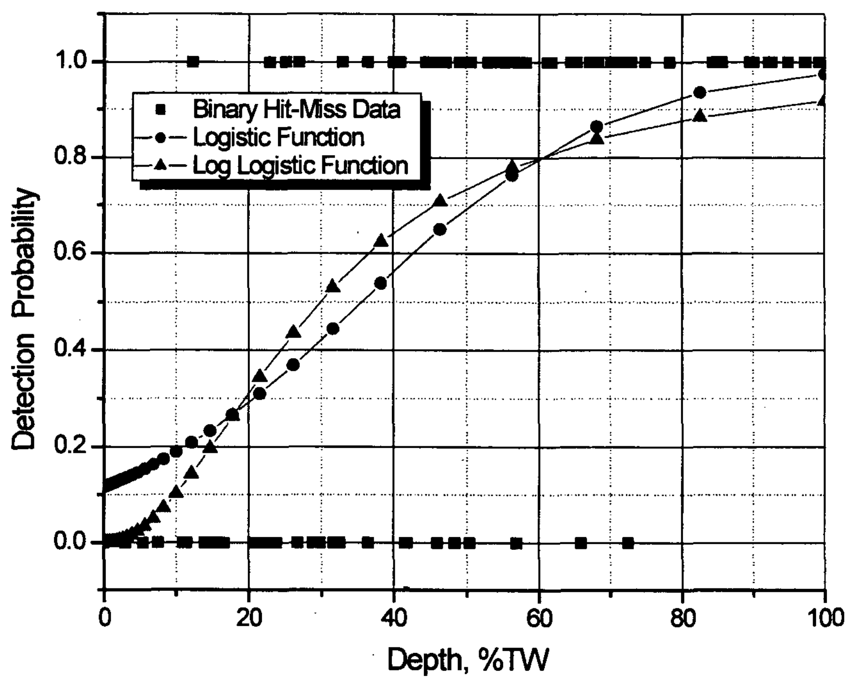

Figure 4-2
Different POD Models Resulting from the Same Hit-Miss Data

4.2.3 GLM Calculation of POD and its Uncertainties

Generalized Linear Modeling (GLM) extends the range of application of linear statistical models by accommodating response variables with non-normal distributions. References [10-11] provide excellent practical introductions to GLM for POD modeling, while References [12-14] discuss mathematical and computational aspects.

The industry approach for addressing POD uncertainties has been defined by the EPRI Tools for Integrity Assessment Project [7-8]. Binary hit-miss data are processed using a GLM spreadsheet shown in Figure 4-3. The spreadsheet provides a POD model, typically a log-logistic function, and confidence bounds for the predicted probabilities.

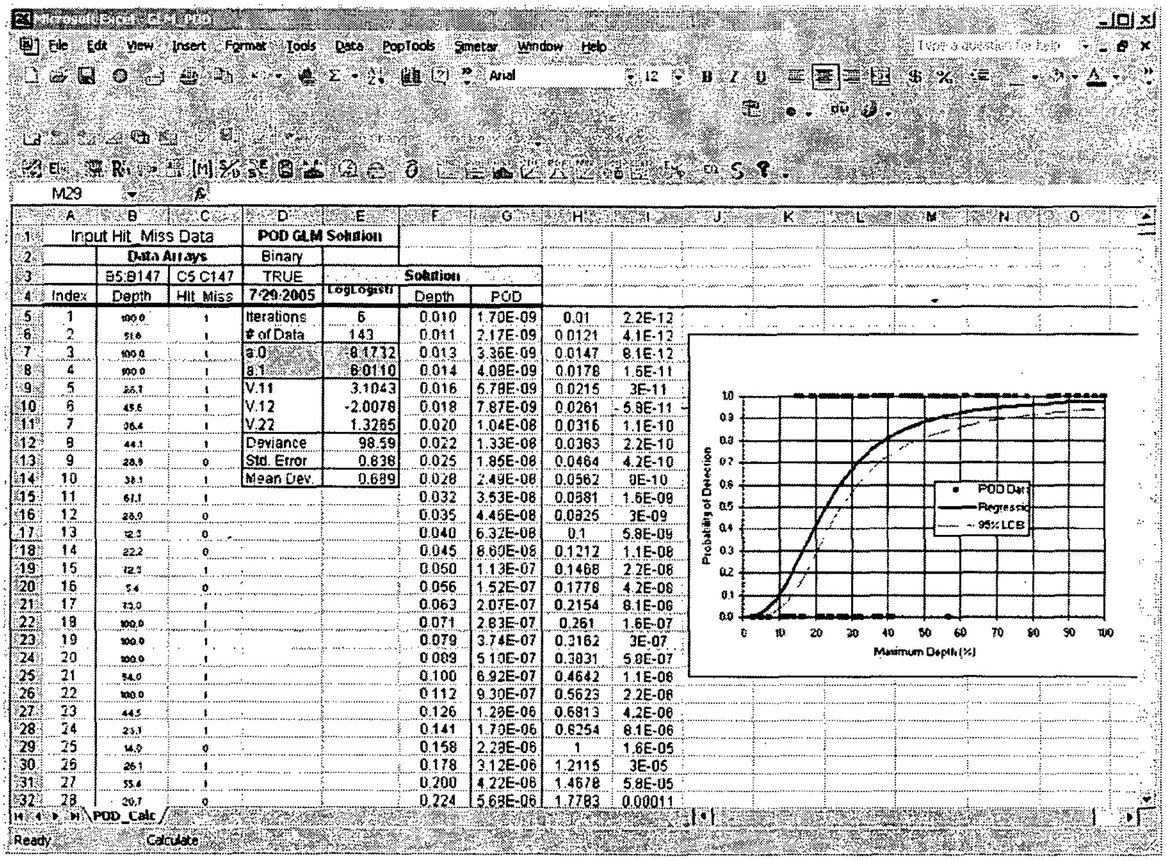

Figure 4-3
GLM Workbook for Calculating POD and POD Uncertainties

The GLM spreadsheet provides a covariance matrix describing the uncertainties in the parameters of the POD model or function. For the case of a two-parameter log-logistic model, with intercept a_0 and slope a_1, the confidence bounds for the predicted probabilities are found using the following expression,

$$a_o + a_1 X_i + Z_p \eta_i, \text{ where } \eta_i = \left[V_{ii} + \left(2V_{12} + V_{22} X_i\right) X_i\right]^{1/2} \qquad \textbf{Equation 4-2}$$

where p is the probability associated with the one-sided confidence level desired, a_0 and a_1 are the intercept and slope values, and the V_{ii} are the covariance matrix values. In using the spreadsheet, the hit-miss data are input, and the POD model or function and the desired confidence bound are selected. After spreadsheet calculation, numerical outputs are provided in three columns consisting of values of the independent variable, the predicted detection probabilities, and lower bound detection probabilities. These results are also plotted for viewing.

4.2.4 Experimental Determination of System POD

System POD captures the effects of procedures, instrumentation and personnel collectively. It can be determined experimentally by having multiple analysis teams analyze a set of NDE data consisting of "grading units" traceable to metallographic ground truth so that the true tube degradation state is known. All testing is done blind. The resulting data analyst detections and non-detections are then coded in binary format i.e., ones and zeros, which, along with ground truth data are entered into the GLM spreadsheet described above for individual analysis team POD calculation.

Figure 4-4 shows multiple analysis team results for an NDE data set constructed using grading units for a particular degradation mechanism and tube examination technique. The spread in individual team performance is noted by the left-most and right-most POD curves. The 95% lower confidence bound (LCB) curve is shown as well as a weighted average POD (GLM Weighted Average). This single system POD and its uncertainty can be derived using a GLM spreadsheet. The weighted-average POD is calculated by: 1) appending all of the multiple team hit-miss results and ground truth data into a single file, and 2) calculating a POD using the conventional GLM spreadsheet described in Figure 4-3. In effect, the composite hit-miss data for a particular grading unit will be "weighted" by the collective team performance rather than a single team. In simple terms it can be viewed as an average POD.

The effects of noise on POD can be determined experimentally using superpositioning or signal injection software as described in [15-16] or a POD adjustment method described in Reference [7]. Superpositioning and injection methods rely on the additive properties of signal and noise in constructing eddy-current grading units for POD determination. The adjustment method utilizes a POD vs. signal to noise transfer function to derive a noise-dependent structural POD. Both methods are empirical requiring eddy-current data sets and multiple analysis teams for POD estimation.

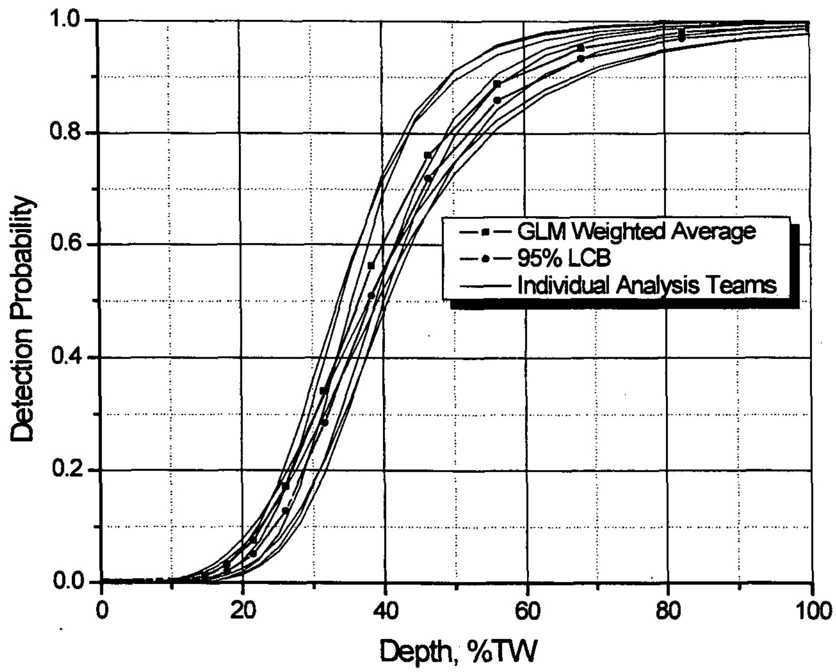

Figure 4-4
Accounting for Data Analyst Uncertainty using a GLM Weighted Average POD

4.2.5 Model-Assisted POD Development

EPRI Model Assisted POD software [17] is a Monte Carlo simulator for predicting a noise-dependent system structural POD for steam generator eddy-current tube integrity assessments. The simulator utilizes a signal response method to describe a paired data set in which the dependent variable is signal amplitude while the independent variable is a structural dimension such as depth, length, or whatever is relevant from a tube integrity perspective. Since signal amplitude and a structural dimension are coupled, the foundation exists for predicting a noise-dependent structural POD. This software was validated [7] by accurately predicting the results of the ODSCC performance demonstration PODs.

The MAPOD software can be used not only to construct PODs, but also to adjust PODs if a change in eddy current noise is determined to negatively affect POD. Figure 4-5 shows two bobbin coil POD models that were calculated for a degradation mechanism located at the center of a support plate and at support plate edges using a MAPOD software. Example data were used to establish a paired data set consisting of voltage and depth which were used as input to the POD simulator along with noise amplitude distributions for the support plate center and edge. The resulting POD for degradation at the plate center is shown in the figure on the far left. The POD for degradation at the plate edge is shifted to the right because of increased noise associated with the plate edges. Additional modeling details are discussed in Appendix D.

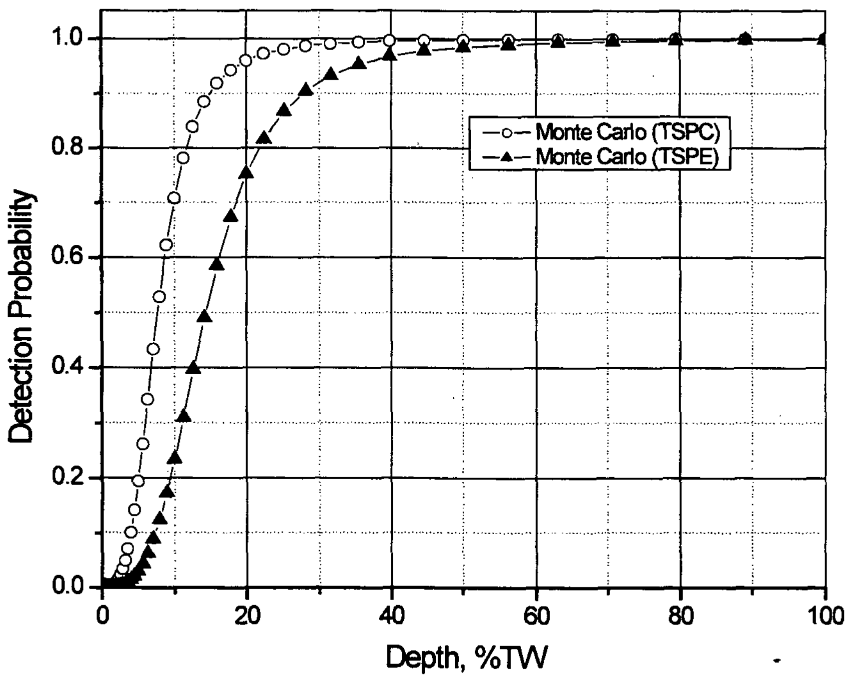

Figure 4-5
Model-Assisted Example POD Calculations Using Data for Volumetric Degradation at Tube Support Plate Center and Edges

4.3 Sizing Requirements and Limitations for Tube Integrity Applications

Performance measures for sizing shall be quantified for the entire NDE system, including the associated uncertainties. NDE sizing uncertainty is based on assessing a measured NDE value with the true value of a structural variable as determined from metallography. Normally this comparison is done over the expected dynamic range of the structural variable, resulting in a paired data set consisting of measured and actual values. Sizing uncertainties are determined using standard linear regression analysis methods.

The SGMP PWR Steam Generator Examination Guidelines [1] provide a protocol for the assessment of sizing capability, and some of the ETSSs provide sizing capabilities in terms of linear regression equations that relate the structural variable used in an integrity assessment to the measured value. The Tools for Integrity Assessment Project has developed sizing correlations for one degradation mechanism (ODSCC) with sizing uncertainties that account for analyst uncertainties as well as technique uncertainties. The ODSCC performance demonstration sizing results show that when sizing ODSCC using amplitude, human factor effects on sizing error is very small. The technique error is on the order of 10-15% through-wall with a human factors component of < 1% through-wall.

The starting point for establishing sizing uncertainties is a scatter plot of NDE measured ($NDE_{measured}$) and true structural (S) values as shown in Figure 4-6. The data shown are taken from an ETSS developed for a PWSCC alternate repair criteria at dented tube support plates and reflect the measurement results for several data analysts. Accordingly, the value of $Sy,x = 14.15$, reflects the system sizing error. The structural variable for this example is maximum depth in percent through-wall. The measured value is used as the independent variable with the structural value being the dependent variable. The paired data are then regressed using the linear equation

$$S = a + b * NDE_{measured}$$ **Equation 4-3**

where S is the structural parameter used for integrity assessment, a and b are constants determined by minimizing the RMS error about the regression line and $NDE_{measured}$ is the NDE measured value. In general, these constants are degradation mechanism and examination technique specific (for Figure 4-6, a=-0.94 and b=0.93). The tube integrity engineer can now determine the structural variable S, which is generally a function of the degradation mechanism being addressed, i.e., thinning, axial cracking, etc. To date, specific values for S have included maximum depth; percent degraded area (PDA); burst effect length and burst effective depth; crack area etc.

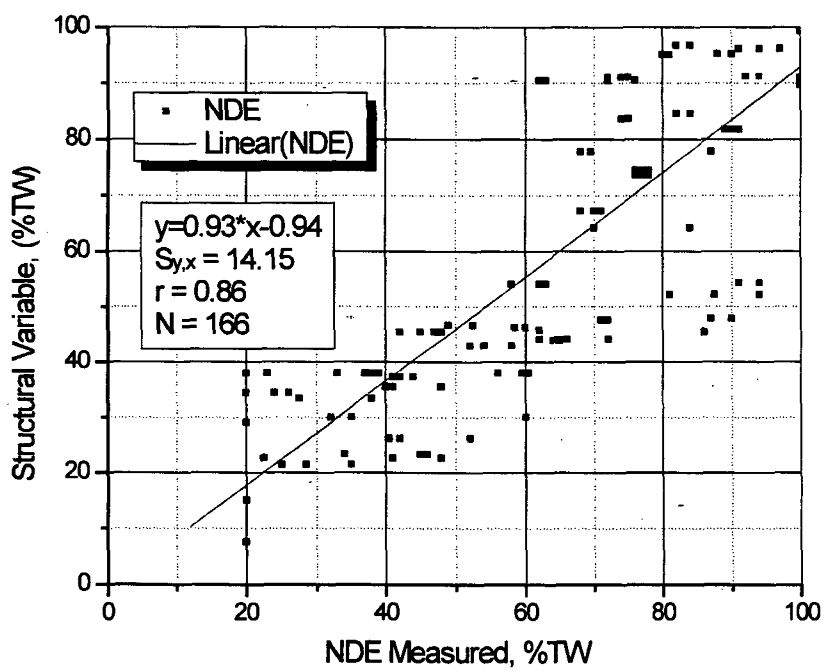

Figure 4-6
Regression Plot Format Used for Determining NDE Sizing Errors – PWSCC ARC Data Set

For correlations that have not been developed in the Tools for Integrity Assessment Project, the sizing data in the ETSSs can be used as a starting point for determining NDE system error values. System values for NDE sizing error can be estimated using ETSS technique values by increasing the standard error of regression $S_{y,x}$ by a factor of 1.12 to account for personnel effects.

The factor of 1.12 is based on the assumption that the analyst uncertainty is 50% of the NDE technique uncertainty. This recommendation is statistically conservative and supported by Reference [18]. If a licensee already has procedures in place that account for system uncertainties, it is acceptable to use them until applicable ETSSs are updated to include system performance indices.

When using the data from the ETSSs or from other data to determine sizing performance, the technique performance test data should have a correlation coefficient that establishes with 95% confidence that a correlation does statistically exist between the measured size and what is considered to be the actual size. Table 4-2 provides the minimum correlation coefficient for a given number of data points [19] to achieve this level of confidence. For example, for a sizing data set of 16 points the correlation coefficient must be greater than 0.5. If the correlation coefficient is less than this, using this data shall be technically justified.

Table 4-2
Correlation Coefficient, r*, at 95% confidence level for a positive correlation [19]

H	Correlation Coefficient r, (minimum)
30	0.36
25	0.40
20	0.44
16	0.50
Note: Minimum number of recommended data points is 16	

* Applies to linear first-order models only

4.4 Extension of Qualified NDE Techniques for Tube Integrity Applications

To satisfy CM and OA, it may sometimes be necessary to extend techniques that are qualified for a particular application to other applications where they have not been formally qualified under Reference 1. For example, if an existing degradation mechanism has no applicable ETSS, a qualified technique would have to be used. When extending techniques, a justification shall be documented.

5
DEGRADATION GROWTH RATES

5.1 Introduction

In this section methods are presented to determine degradation growth rates from NDE inspection information. This section also provides specific growth rate information from industry service experience that may be used when plant-specific data are limited.

Growth rates shall be developed for each existing degradation mechanism identified in the pre-outage reviews discussed in Section 6. Plants that have degradation within their steam generators should analyze inspection data from prior outages to help develop flaw growth information. Flaw growth rates are necessary in order to project EOC degradation distributions when implementing an OA strategy. An empirical growth rate distribution and a 95th percentile upper bound growth rate value should be established for each existing degradation mechanism if sufficient data exist. Data sufficiency requirements are presented below. If sufficient data does not exist for SCC/IGA degradation, the default degradation growth rates supplied in Section 5.5 may be used.

5.2 Background

Growth rate information is obtained from the change of NDE measurement parameters from one inspection to another. For a plug-on-sizing repair scenario, such as for wear, sizing information may be available from current and prior inspections. For a plug-on-detection repair scenario, as is typically applied to crack-like degradation, NDE sizing for the inspection prior to the detection inspection is typically based on a reevaluation of previous NDE data, i.e., a historical or look-back evaluation, using the benefit of hindsight. This naturally raises a substantial question of sizing accuracy, since look-back sizing is applied to an indication that was previously unidentified. This situation can be improved by having the same NDE analyst perform side-by-side sizing of a given indication using the data from both inspections. Growth rates obtained from plug-on-sizing repair scenarios tend to provide the most reliable growth rate data.

Degradation growth rates developed for the DA should be established from past site specific results. If site specific data is not available, then industry data from similar design SGs may be considered for use with the caution that non-site specific degradation rates are typically different due to the differences in SG operation and should be evaluated for application.

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5.3 Data Evaluation Procedures

5.3.1 Conservative Estimate of the Growth Rate Distribution

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5.3.2 Realistic Estimate of the Growth Rate Distribution

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Figure 5-1
Global Average Growth rate of Maximum Depth as Voltage Threshold of Acceptable Sizing Data is Increased

5.4 Illustrations of Estimations of Actual Growth Rate Distributions

5.4.1 Example 1: Wear at Supports, Plug on Sizing

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Figure 5-2
Comparison of NDE Measured Growth Rates, Actual Physical Growth Rates and Computer Simulation of NDE Measured Growth Rates for Wear Depth Growth.

5.4.2 Example 2: Axial PWSCC at Dented Intersections Plug-on-Sizing

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Figure 5-3
Distribution of NDE Measured Average Depth Growth Rates of PWSCC Indications Left In Service under an ARC

Figure 5-4
Distribution of NDE Measured Length Growth Rates of PWSCC Indications Left In Service under an ARC

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5.4.3 Example 3: Axial ODSCC in OTSG Tubes Plug on Detection

5.5 Default Growth Rate Distributions for SCC

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5.5.1 Axial Stress Corrosion Cracking

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Figure 5-5
Comparison of NDE Measured Growth Rates, Actual Physical Growth Rates and Computer Simulation of NDE Measured Growth Rates for Axial PWSCC

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Figure 5-6
Deceleration Factor for ODSCC and PWSCC Growth Rates Compared to 611°F

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Figure 5-7
Comparison of NDE Measured Growth Rate Distribution with Computer Simulation Result of NDE Measured Growth Rate Distribution, OTSG Axial ODSCC/IGA.

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Figure 5-8
Comparison of NDE Measured Growth Rate Distribution with Best Estimate Physical Growth Rates Distribution, OTSG Axial ODSCC/IGA.

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Figure 5-9
Cumulative Distributions of Average Depth Growth Rates of Axial ODSCC/IGA
(curve on the left is a best estimate distribution, others include NDE sizing uncertainties)

5.5.2 Circumferential Stress Corrosion Cracking

6 DEGRADATION ASSESSMENT

6.1 Introduction

Degradation assessment is the process of identifying and documenting existing and potential degradation in planning for an upcoming outage, including inspection plans and related actions for the primary and secondary sides of the steam generator. It is mandatory that a DA be performed prior to each SG inspection and prior to pre-service inspection. The degradation assessment focuses on integrity of the steam generator tubing. Based on past steam generator tube inspection results, the tubing can be subjected to a variety of degradation mechanisms as illustrated in Figures 6-1 and 6-2. The Degradation Assessment along with the OA and operating experience provides the supporting basis for the inspection interval.

6.2 Purpose

The overall purpose of the DA is to ensure that appropriate inspections are performed during the upcoming outage, and that the requisite information for integrity assessment is provided. In conjunction with the steam generator program, the DA shall address the following objectives:

1. Identify existing and potential degradation mechanisms.
2. Identify the limiting structural integrity performance criteria and the appropriate loading conditions for existing and potential degradation mechanisms. (Sections 2 and 3, and Appendix C)
3. Identify CM limit for all existing and potential degradation mechanisms and the appropriate measurement parameter for each degradation mechanism. (Section 3)
4. Select appropriate techniques for detection and sizing and document the NDE measurement uncertainties for each degradation mechanism. Refer to the EPRI PWR Steam Generator Examination Guidelines [1]. (Section 4)
5. Document inspection locations, sampling sizes, and expansion criteria. Refer to EPRI PWR Steam Generator Examination Guidelines [1].
6. Plan for monitoring degradation in tube hardware such as plugs and sleeves and secondary side components such as tube supports and anti-vibration bars.
7. Identify relevant industry operating experience.
8. Identify the secondary side activities planned for the upcoming outage. (Section 10)

Additionally, the DA may be utilized to capture proactive inspections that are performed by the utility that are not required by Technical Specifications or the SGMP PWR SG Examination Guidelines [1]. These inspections can provide useful information to both the utility and the industry. They are typically used to challenge engineering assumptions such as when the Arrhenius equation predicts susceptibility in a future sequential period.

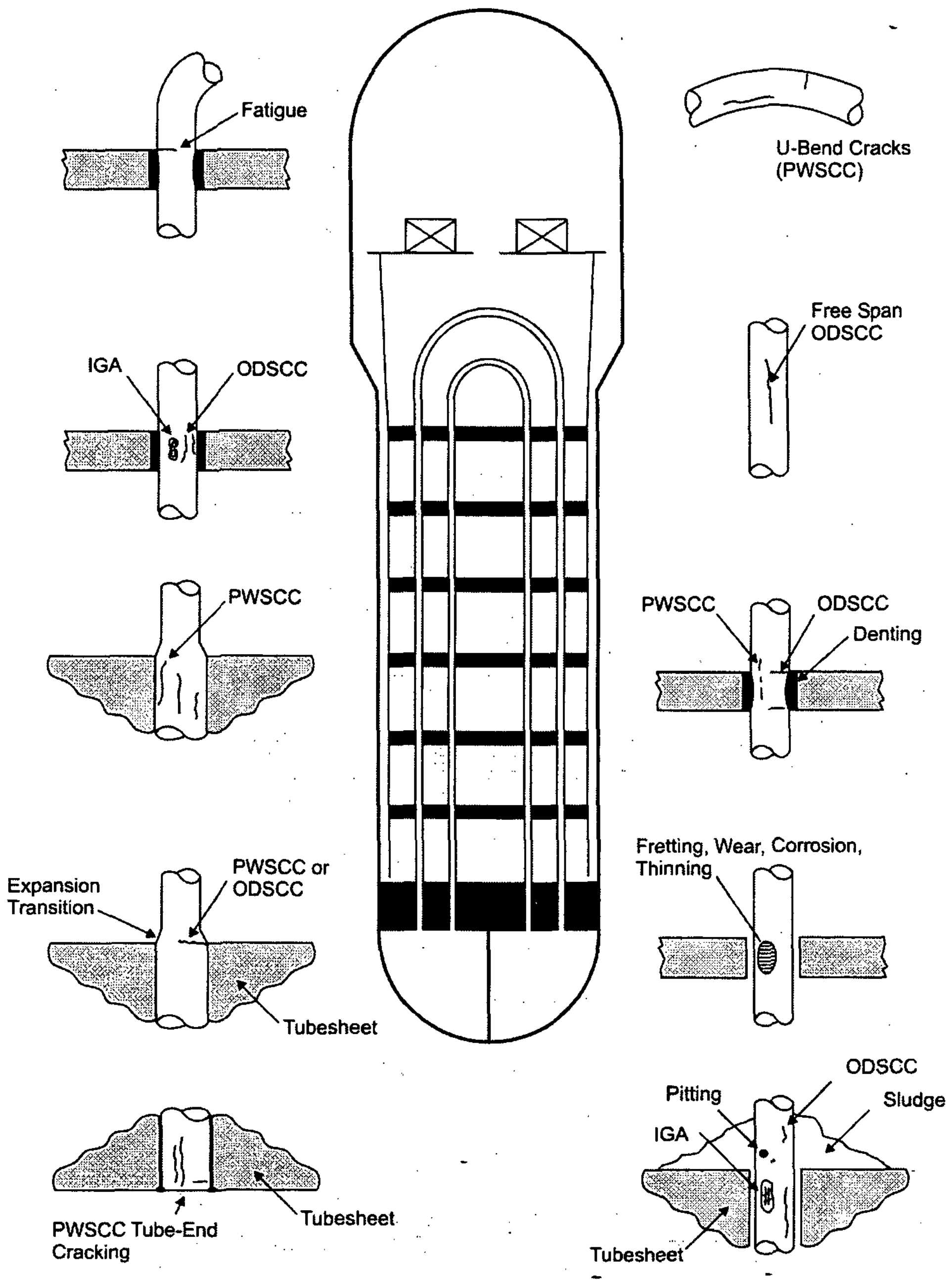

Figure 6-1
Recirculating Steam Generator Degradation Mechanisms

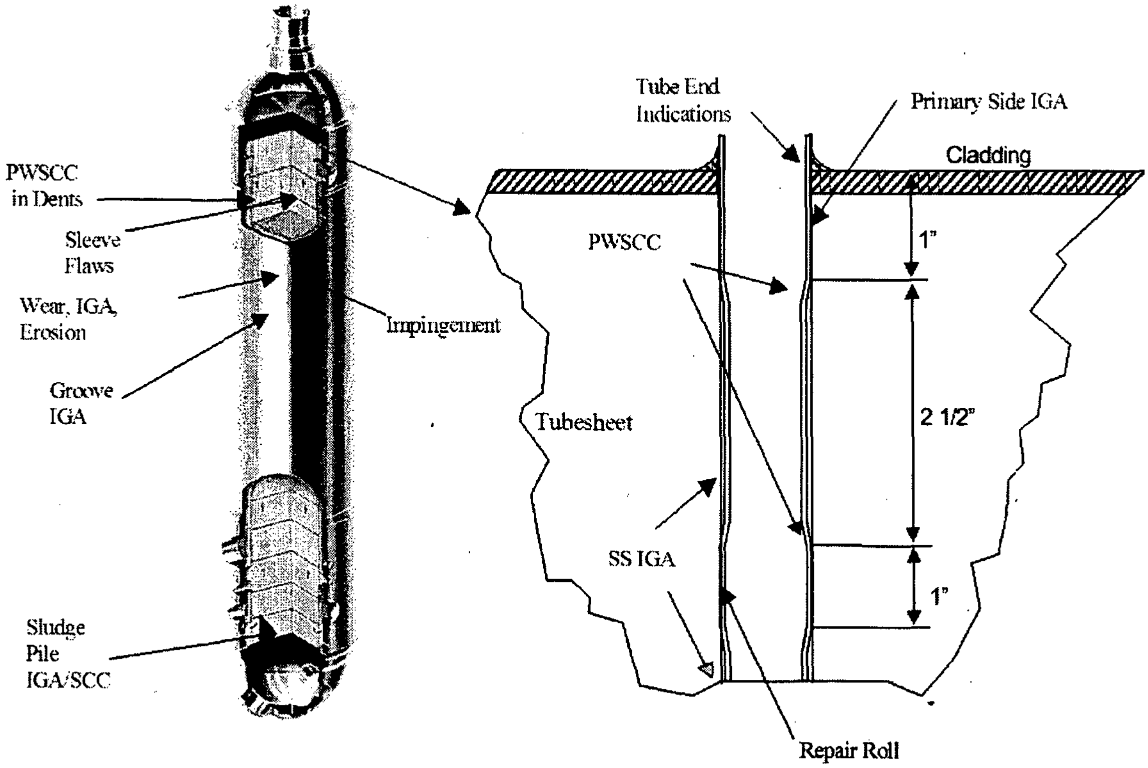

Figure 6-2
Once Through Steam Generator Degradation Mechanisms

6.3 Sources of Information for Degradation Assessment

The Steam Generator Degradation Database (SGDD) **sgdd.epri.com** is an interactive database for reporting and viewing the latest steam generator information. The objectives of this database are to provide a source of information on worldwide SG experiences. It is particularly important to assess degradation found in plants with similar tubing, design, and operating conditions, such as chemistry and temperature.

The EPRI SGMP web site (sgmp.**epriq.com**) includes examination technique specification sheets (ETSSs), plant experiences, eddy current data, links to SGMP guidelines, SGMP Information Letters, SGMP Interim Guidance and other relevant inspection planning information.

Guidance for implementation of NEI 97-06 is provided in the regularly revised SGMP guideline documents. Questions regarding interpretations of the guidelines should be submitted to the Steam Generator Program Review Board using the process outlined in Reference 3. Existing interpretations for the guidelines can be viewed at the SGMP website.

NRC information notices and generic letters provide information and/or requirements and can be found on the NRC web site (**nrc.gov**).

Up to date information can be obtained at the EPRI SGMP Technical Advisory Group meetings.

INPO operating experience and the results of INPO reviews provide pertinent information for input into the DAs. These can be accessed on the INPO web site (**inpo.org**).

6.4 Identification of Potential Steam Generator Degradation Mechanisms

The planning process includes an assessment of potential forms of degradation with identification of their likelihood of occurrence. Anticipating newly developing forms of degradation is necessary for preparation of inspection samples and expansion plans, and identification of applicable NDE equipment, techniques, personnel, and disposition requirements.

A general plant description pertinent to the operation of the steam generator is included or referenced. The description includes details of the SG design, material and manufacturing processes variability. This includes the tubing dimensions, tube sheet and support plate designs, as well as details of the anti-vibration bars. Other design, as built configuration, and plant system information that can affect SG tube integrity is also included in the DA. A listing of similarly designed and/or operated steam generators could prove to be useful. Assessment of potential degradation mechanisms should consider, but not be limited to, experience of steam generators with similar design and materials. Such assessments should consider factors such as the susceptibility of the tubing material to stress corrosion cracking, operating temperature and pressure, residual stresses associated with tube bending or expansion processes during fabrication or operation, and secondary side water chemistry, including the potential for contaminants such as lead, chlorides, sulfates, and copper. It is important for DAs to consider that many degradation mechanisms have occurred due to deviation of conditions from nominal.

To provide appropriate outage planning, prior to a refueling outage or in an outage where SGs will be inspected, previously identified degradation mechanisms on both the secondary and primary sides of the steam generator that affect tubing, support structures, pressure and leak boundaries should be characterized as to location and possible extent. Growth rate information for the degradation forms should also be reviewed, if available. The plugging and repair status should be discussed. It is also useful to include preventive measures that have been taken such as heat treatment, chemical cleaning, sludge lancing, or shot peening. If tubes have been pulled, these should be identified and results of the examinations discussed.

6.4.1 Degradation in Previously Plugged Tubes

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6.4.2 Types of Degradation

Tube degradation in steam generators can be due to corrosion mechanisms, mechanical wear, and fatigue. Further, corrosion degradation may develop at locations of local discontinuities such as dings, dents, scratches or other mechanical marks. Typical locations and forms of degradation are shown in Figure 6-1 and 6-2, and detailed discussions can be found in Reference [5]. The following is a listing of potential degradation mechanisms and local discontinuties. It is not intended to be an all inclusive list. Only the types of degradation mechanisms applicable to the SG and its tubing need to be addressed in the DA.

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6.4.2.1 Intergranular Attack and Outside Diameter Stress Corrosion Cracking

6.4.2.1.1 Outside Diameter Stress Corrosion Cracking at Dents and Dings

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6.4.2.2 Primary Water Stress Corrosion Cracking and Intergranular Attack

6.4.2.2.1 Primary Water Stress Corrosion Cracking at Dents and Dings

6.4.2.3 Tube Fretting and Wear

6.4.2.4 Other Wear Damage

6.4.2.5 Pitting

6.4.2.6 High Cycle Fatigue

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6.4.2.7 Impingement

6.4.2.8 Wastage/Thinning

6.5 Identification of NDE Techniques

NDE techniques selected for detection and/or sizing of degradation in tubing, including tube repairs shall be selected based on the assessment of existing and potential degradation. Information required for integrity assessment such as POD and sizing uncertainties shall be identified (See Section 4). Inspection requirements are specified in Section 3 of the SGMP PWR Steam Generator Examination Guidelines [1].

6.6 Identification of Inspection Sample Plan

An appropriate tube bundle sampling strategy and expansion plan shall be defined for each degradation mechanism identified following the guidance in the SGMP PWR Steam Generator Examination Guidelines [1]. Discrete sample plans shall be documented [1].

6.7 Integrity Assessment and Repair Limits

Pre-outage preparations begin with identifying SG degradation forms as discussed above. The limiting structural integrity performance criteria and the appropriate loading conditions for the degradation of interest must be determined. In general, based on currently applicable structural performance criteria, this will usually be an applied differential pressure, ΔP, equal to three times the normal operating primary-to-secondary pressure difference, i.e., $3{\cdot}\Delta P_{NOp}$ to demonstrate adequate resistance to burst. Bending loads may also be important. There may be circumstances where the combination of degradation mechanism, geometry and material properties leads to other applicable criteria (see Section 3 and Appendix C).

Condition monitoring limits for structural and leakage integrity shall be established, identifying the uncertainties associated with the essential variables of integrity evaluation, and documented for each existing and potential degradation mechanism (Sections 3 and 9). These limits can be used to set up in situ screening criteria.

The Technical Specification repair limit is typically 40% through wall when adequate sizing capability exists for the indications. Through review of plant specific, or industry typical defect growth rates, the appropriate tubing repair limits can also be calculated or identified. This will identify which degradation mechanisms are adequately covered by the plant's present Technical Specification repair limit. Repair on detection is the strategy employed for all cracks and volumetric indications where no acceptable sizing capability is available. Alternate repair criteria allow tubes with flaws to remain in service for specific kinds and location of indications. Preventive repair may be required based on OA considerations. Depending on the growth rate of the degradation mechanism(s) that is identified from the inspection, and the length of time between planned inspections (multi-cycle inspection intervals), it may be necessary to establish a lower through-wall limit for plugging to ensure tube integrity at the time of the next inspection.

When a tube that is plugged has the potential to sever, stabilization criteria are used to decide whether or not to stabilize the plugged tube (see Section 6.4.1). In this situation, each plant shall establish appropriate criteria.

6.8 Secondary Side Considerations

The Degradation Assessment shall include secondary side considerations (see Section 10).

6.9 Actions upon Finding Unexpected Degradation

When an unexpected degradation mechanism is found during the steam generator inspection that was not addressed in the current DA, the mechanism shall be entered into the plant's corrective action program. The corrective actions shall include the inspection requirements for the degradation to support the tube integrity assessments. Reporting should be in accordance with Section 11.

6.10 Actions for New Operating Experience When the Degradation Assessment is Complete and Approved

The potential exists for the industry to issue new applicable operating experience after the unit's DA has been approved for the upcoming/present outage. If practical, the DA should be revised to reflect the operating experience and the outage plans adjusted as required. In the event that circumstances do not lend to a timely consideration of the operating experience in the DA, such as a lengthy approval process, the operating experience should be captured and evaluated in the plant's corrective action program. The evaluation should consider the safety significance for the unit and identify any appropriate actions, such as additional inspections or inspection techniques needed, for the upcoming/present outage and any impact on operation until the next scheduled inspection.

6.11 Review of the Degradation Assessment prior to a Refueling Outage without SG Inspections

A review of the DA and OA information used in evaluating tube integrity shall be performed prior to each refueling outage when SG primary and/or secondary-side inspections are not scheduled to validate the inspection interval. See section 11.2.4 for additional review requirements.

7
CONDITION MONITORING

7.1 Introduction

Condition monitoring (CM) involves the evaluation of inspection results at the end of the inspection interval to determine the state of the steam generator tubing for the most recent period of operation relative to structural and leakage integrity performance criteria. This section provides guidance on performing structural assessments. Guidance for leakage assessments is given in Section 9. Condition monitoring can be accomplished by analytical methods or by in situ pressure testing. The CM evaluation is required by NEI 97-06 [2].

Condition monitoring provides a means of verifying EOC tube integrity and previous OA predictions. This guideline identifies a computational hierarchy for combining uncertainties in these assessments when structural and leakage integrity is inferred from NDE inspection results. The assessment strategies include an arithmetic bounding approach, a simplified statistical approach, and Monte Carlo analysis. In order to decide on the appropriate evaluation methodology, information provided in the following discussion specifies the essential elements of the different assessment strategies. In situ pressure testing may be needed to demonstrate structural and leakage integrity. All detected degradation shall be evaluated, including secondary side inspection results, for structural and leakage integrity.

7.2 Condition Monitoring Evaluation Procedure

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7.3 Structural Integrity Evaluation Using Inspection Results

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7.3.1 Probabilities and Percentiles

7.3.2 Arithmetic Strategy for Combining Uncertainties

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7.3.3 Simplified Statistical Strategy for Combining Uncertainties

7.3.4 Monte Carlo Strategy for Combining Uncertainties

Table 7-1
Condition Monitoring Uncertainty Treatment for Structural Integrity

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7.3.5 Strategy Comparison

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7.3.5.1 Arithmetic Evaluation

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7.3.5.2 Simplified Statistical Evaluation

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7.3.5.3 Monte Carlo Evaluation

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Figure 7-1
Condition Monitoring Structural Limit Curves for Axial PWSCC Per ETSS 96703.1 at 4155 psi Using Three Strategies for Combining Uncertainties

7.4 Signal Amplitude Approaches to Structural Integrity

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Figure 7-2
Condition Monitoring Plot for Freespan Axial ODSCC/IGA in OTSG Tubing at 4050 psi

7.5 Role of In Situ Pressure Testing

7.6 Verification

A comparison of the CM results to the previous cycle OA predictions shall be performed. If the previous cycle OA did not bound the CM results, a root cause evaluation, in accordance with the Licensee Corrective Action Program, shall be performed to identify the reason and the applicable values in the forthcoming OA shall be changed to account for the difference. This verification process is also part of the OA process for the next cycle of operation and is discussed further in Section 8.

If, upon completion of the CM evaluation, which may include in situ pressure testing, the results indicate that structural and/or leakage integrity evaluations fail to satisfy any of the performance criteria, the condition shall be reported to the NRC and Industry in accordance with the reporting requirements of the Licensee's Technical Specifications and NEI 97-06 [2]. Failure to meet the CM criteria indicates that the conclusions of the prior OA were incorrect. Therefore, a root cause evaluation shall be completed and necessary corrective actions shall be identified.

Some examples of appropriate corrective actions include: lowering the repair limit to account for unexpectedly high degradation growth rates; reanalysis of eddy current data to increase detection sensitivity; augmentation of eddy current inspection with an alternate NDE technique such as UT; and reducing inspection interval. In some cases, identified corrective actions may require an extended period of time to implement. For example, a tube may have to be pulled to identify the nature and severity of degradation.

8
OPERATIONAL ASSESSMENT

8.1 Introduction

Operational Assessment (OA) involves projecting the condition of the SG tubes to the time of the next scheduled inspection outage and determining their acceptability relative to the tube integrity performance criteria of NEI 97-06 [2]. All detected degradation mechanisms shall be evaluated, including secondary side inspection results. Forms of degradation that have been found at prior inspections but have not been observed at the current inspection shall also be evaluated.

The purpose of this section is to provide guidance for performing an OA and evaluating the results. The focus of this section is structural integrity. Leakage integrity is covered in Section 9.

Intervals between inspections of the SGs depend on the results of the OA. NEI 97-06 [2] requires that an OA be performed after each SG inspection. The intent is to assess tube integrity for inservice degradation in operating SGs. In general, for new or replacement SGs, an OA for the interval between baseline and the inspection after the first cycle of operation is not required. An example when an OA should be performed for new or replacement SGs for initial operation would be when baseline inspections detect foreign objects which can not be retrieved. In this example, an OA would assess the potential for tube wear during the first inspection interval.

The fundamental objective of an OA is to ensure that structural and leakage performance criteria will be met over the length of the upcoming inspection interval. It shall be demonstrated that the degradation detection sensitivity and/or NDE sizing uncertainty combined with degradation growth rates leads to the expectation that structural and leakage integrity criteria will be met at the end of the next inspection interval. In terms of structural integrity, the fundamental OA requirement is that the projected worst case degraded tube for each existing degradation mechanism shall meet the limiting structural performance parameter with a 0.95 probability at 50% confidence.

During actual operation of a given SG in a given cycle, for the degradation mechanism of interest, one tube has the lowest structural performance parameter associated with it. However, this is one of many possible outcomes for the given starting condition. A fully probabilistic analysis leads to a distribution of possible outcomes of SG operation for all degradation sites both detected and undetected for a given degradation mechanism. An illustration of a fully probabilistic treatment of the tube bundle for a given mechanism is shown in Figure 8-1. In this illustration, a Monte Carlo analysis for the tube bundle is used to establish the distribution of the tubes that produce the lowest burst pressure for the given degradation mechanism. As shown in Figure 8-1, the "projected worst case degraded tube" is defined as the degraded tube with the 5th

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Figure 8-1
Fully Probabilistic Monte Carlo Simulation to Established Worst-Case Degraded Tube – Full Bundle Analysis

8.2 Projection of Worst Case Degraded Tube

The main objective of an OA is to demonstrate that any inservice tube degradation will not challenge tube integrity as defined by the structural performance criteria for tube burst and accident induced leakage. Therefore, the determination of the worst-case degraded tube until next inspection is an important aspect of the OA.

8.2.1 General Methods and Strategies

There are two general approaches to the projection of the worst case degraded tube, direct calculation and use of simplified techniques. Direct calculation involves consideration of the entire flaw population for a given degradation mechanism and associated procedures to project the distribution of EOC extreme value structural integrity parameters so that the 5^{th} percentile value may be obtained. Direct calculation methods are referred to herein as fully probabilistic. However, most OAs can be performed using less complicated approaches to project the worst case degraded tube at the end of the inspection interval of interest. Simplified methods of projecting the worst case degraded tube are designed to provide conservative approximations of fully probabilistic calculations that consider the entire projected flaw population and the variety of possible outcomes for a given cycle of operation. In the simplified techniques, the worst case flaw is projected using conservative assumptions coupled with uncertainties that are combined using the Arithmetic, Simplified Statistical or Monte Carlo calculation strategies. Depending on available information and margins, the OA strategy may range from a simplified approach to a fully probabilistic analysis.

It is important to note that there are two basic repair strategies or options for detected degradation, Repair on Detection and Repair on NDE Sizing. Repair on Detection means that no detected degradation is intentionally left in service, hence, the basic issue for performing the OA is to account for the undetected flaw population. Estimating the beginning of cycle flaw does not include NDE sizing uncertainties because nothing is being measured. If a less than 100% inspection is being performed and degradation is detected, the OA shall consider the uninspected population. If degradation is identified in a less than 100% inspection, one cannot assume that the worst case flaw has been identified. For a Repair on NDE Sizing scenario, detected degradation is intentionally left in service if the size determined by NDE is below a certain level. Practically, this level is the technical specification repair limit, typically 40%TW NDE maximum depth, although some plants have slightly higher NRC-approved technical specification repair limits. The most important considerations in this scenario are accounting for the NDE sizing uncertainties of the flaws left in service and a conservative growth allowance.

The 40% TW NDE maximum depth repair limit is usually only applied to tube wear at supports and cold leg thinning degradation at tube supports. Other types of degradation may remain in service according to NRC approved Alternate Repair Criteria (ARC). Some examples of ARCs are:

- Bobbin Voltage ARC for Axial ODSCC/IGA at Drilled Tube Support Plate Intersections
- Depth Based +Point Sizing of Axial PWSCC at Drilled Tube Support Plate Intersection

- Bobbin Based Depth Sizing of First Span Volumetric IGA in an OTSG Plant
- Various Tubesheet Crevice Degradation ARCs where SLB Leakage is the Limiting Issue since Burst is Prevented by the Presence of the Tubesheet.

Degradation covered by NRC approved ARCs require the use of prescribed methodologies that are not presented in this document.

While CM deals with observed NDE measurements of degradation and whether the structural and leakage resistances of the tubes meet the desired performance criteria, OA typically deals with the projected actual physical states of the degradation and is typically performed by:

- Determining degradation dimensions at BOC,
- Adding a growth allowance to determine the EOC degradation dimensions, and,
- Evaluating the projected EOC dimensions versus the limiting structural and leakage integrity performance criteria, including uncertainties in burst/strength equations and material properties.

Section 8.3 describes several approaches to performing an OA where the above three steps are implemented in a fully probabilistic manner considering the entire flaw population for a given degradation mechanism. These descriptions are qualitative since fully probabilistic approaches are usually only implemented in special circumstances. Most often, the projection of a worst case degraded tube for each degradation mechanism can be accomplished in a conservative fashion using less complicated computational schemes. Several simplified methods and their general applicability are described next.

8.2.2 Simplified Analysis Methods

Relatively simple schemes are presented in detail herein, with examples to clarify the calculation procedures. These methods involve a single tube analysis that will provide a conservative estimate for the projected EOC lower 95th percentile extreme value burst pressure in most cases. Repair on NDE Sizing is covered first followed by Repair on Detection. Uncertainties in the single worst case flaw projection schemes are combined by one of three strategies:

- Arithmetic
- Simplified Statistical
- Monte Carlo

Uncertainties may be combined using a mix of the analysis strategies. Procedures employing a mix of strategies are included herein with examples. Results of different assessment strategies are compared showing that increased calculational effort may be needed to increase accuracy and reduce conservatism. Since OA deals with projected degradation states, comparison of projections with the CM observations is important to verify that the selected OA methodology and input parameters are appropriate. The importance of various parameters in this verification process, such as flaw severity and numbers of flaws, is described. Tables outlining the calculation steps needed to project worst case degraded tubes using the three assessment strategies for combining uncertainties are included in what follows.

Two final general points are important:

- To assist the OA process, it is recommended that a facility specific checklist that outlines the steps necessary for successful implementation be developed, depending on the design and condition of the SGs.
- Prior to Mode 4, a review of the CM results and growth shall be performed. If CM results are not as expected and/or growth rates are larger than expected, one or more tubes may fail to satisfy the performance criteria prior to the next scheduled inspection. In this case, remedial actions can be taken. For example, the length of the inspection interval and/or the tube repair criteria could be adjusted. Waiting for the final OA report after the next inspection interval has begun limits options for remedial actions if needed.

8.2.3 Applicability Limits on Simple Assessment Methods

As mentioned earlier, the simplified statistical methods (i.e., arithmetic, simplified statistical and simplified Monte Carlo) are approximate techniques that will yield conservative results for the projected worst-case degraded tube parameters in most situations. The convenience of the simplified procedures, as described Section 8.4 and 8.5, is desirable for situations in which structural integrity parameters are not significantly challenged by existing mechanisms.

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8.3 Fully Probabilistic Operational Assessment Methods

Fully probabilistic OA approaches require the use of well developed computer codes, and the level of experience and effort required to correctly perform this calculation can be significant. However, these approaches can provide definitive and accurate projections of detected and undetected flaw populations. Both Repair on Detection and Repair on Sizing strategies are described. A fully probabilistic approach is sometimes referred to as a full-bundle analysis. The terms are meant to imply that all degradation sites in the bundle for a given degradation mechanism are included in the analysis.

8.3.1 Repair on Detection

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8.3.2 Repair on NDE Sizing

8.4 Simplified Analysis Procedures for Repair on NDE Sizing

The following five sections describe and illustrate simple methods to project the EOC worst case degraded tube and its associated structural integrity for a Repair on NDE Sizing strategy while accounting for all appropriate uncertainties. The various uncertainty strategies and procedures for their use are summarized in Table 8-1. Operational Assessment structural integrity is demonstrated if the projected worst case degraded tube meets the limiting structural integrity performance criterion with at least a probability of 0.95 with 50% confidence. Undetected flaws in a Repair on NDE Sizing strategy can and should be included if the applicable POD curve indicates that undetected flaws and/ or flaws initiating during the cycle of operation are an issue. If undetected flaws and/or flaws initiating during the cycle of operation are an issue, the simple analysis techniques listed in Table 8-1 are not applicable.

Table 8-1
Operational Assessment Uncertainty Treatment for Structural Integrity for Repair on NDE Sizing

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Table 8-1 (continued)
Operational Assessment Uncertainty Treatment for Structural Integrity for Repair on NDE Sizing

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8.4.1 Arithmetic Strategy for Repair on NDE Sizing

8.4.2 Simplified Statistical Strategy for Repair on NDE Sizing

8.4.3 Mixed Arithmetic/Simplified Statistical Strategy for Repair on NDE Sizing

8.4.4 Monte Carlo Strategy for Repair on NDE Sizing

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8.4.5 Strategy Comparison for Repair on NDE Sizing

8.4.5.1 Example: Cold Leg Thinning at Drilled Tube Support Plates

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Figure 8-2
Cumulative Distribution of Cold Leg Thinning Depth Growth Rate NDE Measurements, Computer Simulation of NDE Measurements, and Best Estimate Growth Rate Distribution

8.4.5.2 Arithmetic Strategy

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8.4.5.3 Mixed Arithmetic/Simplified Statistical/Monte Carlo Strategy

8.4.5.4 Simplified Statistical Strategy

8.4.5.5 Monte Carlo Strategy

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8.5 Simplified Analysis Procedures for Repair on Detection

Table 8-2
OA Uncertainty Treatment for Structural Integrity for Repair on Detection

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Table 8-2 (continued)
OA Uncertainty Treatment for Structural Integrity for Repair on Detection

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8.5.1 Arithmetic Strategy for Repair on Detection

8.5.2 Simplified Statistical Strategy for Repair on Detection

8.5.3 Mixed Arithmetic/ Simplified Statistical Strategy for Repair on Detection

8.5.4 Monte Carlo Strategy for Repair on Detection

8.5.5 Comparison of Strategies for Repair on Detection

8.5.5.1 Example Equation

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8.5.5.2 Arithmetic Strategy

8.5.5.3 Mixed Arithmetic/Simplified Statistical Strategy

8.5.5.4 Simplified Statistical Strategy

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8.5.5.5 Monte Carlo Strategy

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8.6 Verification

8.7 Review of the Operational Assessment prior to a Refueling Outage without SG Inspections

Prior to a refueling outage that does not have planned SG primary-side and/or secondary-side activities, the information used in projecting SG tube integrity in the OA shall be reviewed. There may have been subsequent plant or industry experience that impacts the information used in the tube integrity assessment process that might impact the ability to have an outage without SG inspections. See Section 11.2.4 for additional review requirements.

9 PRIMARY-TO-SECONDARY LEAKAGE ASSESSMENT

9.1 Introduction

This section provides requirements for primary-to-secondary leakage assessment and documents methods to calculate leakage. For CM, degradation detected during an inspection shall be evaluated against the accident-induced leakage performance criterion. Degradation length or depth measured for CM purposes is adjusted for NDE measurement uncertainties. Leakage at normal operating conditions is monitored during plant operation and shall be compared to the operational leakage performance criteria. Operational assessment shall be performed to provide assurance that the leakage integrity performance criteria will be met until the next scheduled SG inspection. Degradation length or depth estimated at the EOC is not a measured parameter and therefore no NDE measurement uncertainties need be applied.

9.2 Accident Induced Leakage

The allowable tube leakage limit is defined by the accident leakage performance criterion in Section 2. Leakage limits shall be met for all design basis accidents, other than a steam generator tube rupture, and shall not exceed the leakage assumed in the plant accident analysis in terms of the total leakage for all steam generators and the leakage rate for an individual steam generator. The maximum leakage limit is further limited to not exceed 1 gpm per steam generator unless an approved specific alternate repair criteria is being implemented. Consequently, it is useful to identify the limiting accident for leakage. The limiting accident may depend on the type of tubing degradation of interest, for example, high axial loads are significant for circumferential degradation but not for axial degradation. The limiting accident is defined by the combination of accident specific loads and the accident specific leakage limit leading to the smallest allowable flaw size. It is this flaw size that must meet CM and OA requirements.

Typically, the limiting accident for leakage is simply the accident producing the largest tube loads. However, this may not always be the case. There may be accidents with a low allowable leakage limit combined with loads that, while less than the maximum, lead to the smallest allowable flaw size for leakage. Accident specific loads and accident specific leakage limits must be evaluated to identify the limiting leakage accident. This can be a difficult exercise since leakage limits based on dose assessments must be combined with accident loads that may be grouped under umbrella transients for convenience and economy. In the absence of more detailed information it is conservative to construct a bounding case by combining the lowest allowable accident leakage with the largest accident tube loads. For plants with accident analyses that assume the same accident leakage for all design basis accidents, other than steam generator tube rupture, the limiting accident for leakage is not necessarily the accident producing the largest tube loads. Prior to each outage, the limiting leakage accident and allowable leakage value shall be confirmed.

Several plants have made commitments to the NRC Regulatory Guide 1.183 [30] which provides an alternate source term approach for the parameters and assumptions to be met for accident analysis. This would enable plants to increase accident induced leakage limits. These limits require approval by the NRC prior to implementation.

This section provides recommended approaches for calculating both leakage through cracks in steam generator tubes and flaw sizes leading to 100 %TW throughwall penetration under combined accident loads and thus accident leakage contributions. Applications to CM and OA leakage integrity evaluations are discussed. Note that the effect of contributing primary loads other than pressure and axial secondary loads that must be treated as primary loads in OTSG's shall be included in leakage integrity evaluations. In practice this reduces to consideration of axial tensile and bending loads when evaluating circumferential cracking and the circumferential extent of volumetric degradation.

9.3 Operational Leakage

The allowable operational leakage limit is defined in plant Technical Specifications and Section 2. Primary-to- secondary leakage that develops during operation shall be evaluated per the latest revision of the PWR Primary-to-Secondary Leak Guidelines [31].

The following information generalizes the relationship between operational and accident-induced leakage limits and is provided by the NRC in Regulatory Issue Summary 2007-20 [32]:

> The loading conditions on the tubes during an accident may be different than the loading conditions on the tubes during normal operation. As a result, the primary-to-secondary leak rate observed during normal operation may change under accident conditions. In some cases, the primary-to-secondary leak rate may increase as a result of the accident, while in other cases it may decrease. If the loading conditions during an accident result in an increase in the primary-to-secondary leak rate (when compared to the normal operating leak rate), it may be necessary to restrict the normal operating leak rate to less than the normal operating leakage rate limit. This applies not only to units that assume the primary-to-secondary leak rate observed during the accident is the same as the normal operating primary-to-secondary leak rate limit, an assumption that is permitted by the NRC's Standard Review Plan; but also to other units since the increase in primary-to-secondary leak rate going from normal operating conditions to accident conditions can result in significant increases in the leak rate (depending on the accident). The actual amount that the leak rate may increase is a function of several factors including the type of flaw that is leaking. For example, the leak rate from a crack may increase significantly (e.g., by an order of magnitude depending on through-wall crack length) under accident conditions [32].

If operational leakage causes a forced outage, a root cause evaluation shall be performed and included as part of the OA report for the forced outage. A forced outage can result from incorrect assumptions or errors in past analyses.

If operational leakage is less than shut-down levels and is consistent with that predicted by the OA, no adjustments to OA methodologies are required; however in situ pressure tests may be required. If operational leakage is not predicted by the OA, assessment strategies shall be modified accordingly.

During an inspection outage following operational leakage of greater than 5 GPD in any SG, the following steps shall be taken to establish information about the leak:

1. Determine which SG(s) are leaking: Monitor all SGs to determine which SG(s) are leaking.
2. If possible, determine the source of the leakage: This is typically performed by a hydrostatic test, bubble test, or helium leak test to identify suspect tube(s) locations on the tubesheet. Quantify the rate (for example, drops per minute or gallons per minute [liters per minute]) of leakage. Correlate the calculated leakage (pressure/temperature adjusted leakage) versus the operational leakage. Determine if results have accounted for the observed operational leakage, while recognizing that an accurate comparison of operating and shutdown leakage measurements is difficult. If the source of the leakage cannot be identified using the methods described above, 100% eddy current examination should be considered. If the eddy current examination locates the potential leakage, proceed with Step 4. If the leakage has not been identified, an evaluation of the actions within Step 6 should be considered.
3. Examine leaking location(s): This inspection is typically performed by bobbin coil eddy current examination to establish axial location within the SG.
4. Examine to determine extent, orientation, and morphology: This is typically performed by rotating coil or array coil technology. Refer to the SGMP PWR Steam Generator Examination Guidelines [1].
5. Review prior inspection history: Review the information contained in the database and the actual historical bobbin and rotating data to establish factual information about the data. If the leakage is originating from a plug or sleeve, review the installation records for that location. Evaluate if installation parameters were met and identify any inconsistencies or nonconforming conditions.
6. Perform a root cause evaluation that includes all SG program elements in accordance with the utility's program(s). This evaluation should address the need to perform eddy current and/or secondary-side visual inspections. Also consider supplementing the root cause team with industry peers. The root cause team shall identify immediate, short-term, and long-term actions to correct any process deficiencies.
7. Execute root cause corrective actions
8. Update and revise the DA, CM, and OA as necessary to address the unexpected leakage.
9. Perform required repairs.

9.4 Leak Rate Calculation Methodologies

The purpose of this section is to document methods that can be used to calculate leakage through cracks in steam generator tubes. These methods are based on several industry leakage assessments using available analytical tools and test data developed by EPRI [27, 33, 34].

Leakage integrity analyses use NDE flaw size information for CM (identified flaws) or OA projected flaw by assessing either those defects which have peak depths near 100% through wall (TW), or those partial depth defects which satisfy structural integrity only by virtue of having lengths less than the critical crack length, but can "pop-through" under accident pressure loadings to create a leakage path.

9.4.1 Leak Rate Equations

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9.4.2 Crack Opening Area Calculations

9.4.2.1 Axial Cracks

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9.4.2.2 Circumferential Cracks

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Figure 9-1
Crack Opening Area versus Axial Stress, Bending Transient Followed by Build up of Axial Tensile Load

Alternative methods of leak rate calculations may be applied if preferred to the approach of Section 9.4. The key consideration is verification of both the flow methodology and crack opening area calculations. Elastic crack opening areas are usually well predicted by a variety of formulations in the literature but plasticity corrections are approximate and need to be verified experimentally.

9.5 Validation of Leak Rate Calculations

Experimental data exists for flow through cracks under reactor operating and postulated accident conditions. The data show considerable scatter but the leak rate methodology of Section 9.4 provides good bounding results. One comparison with available data is shown in Figure 9-2 [33]. The dotted lines represent the leak rate calculations for axial cracks in 0.750 inch and 0.875 inch diameter tubing. The heavy solid line represents data for fatigue cracks. Other data from fatigue cracked specimens are shown as solid diamonds. Finally, data on laboratory stress corrosion specimens are shown as open square symbols.

As expected, smooth fatigue cracks exhibit higher leak rates than stress corrosion cracks of the same axial through wall length. Both the fatigue cracked and stress corrosion cracked specimens exhibit minimal, as cracked, crack opening areas. Care must be taken in interpreting the validity of benchmarking leak rate data. Laboratory produced cracked specimens, either fatigue cracked or stress corrosion cracked, can be mistakenly blunted, which results in large crack opening areas which are not representative of cracks produced in service.

Calculated leak rates are shown to be comparable to measured leak rates for specimens with sharp fatigue cracks. Fatigue cracks produce a conservative upper leak rate compared to service produced cracks. The calculated leak rates are somewhat lower than the measured leak rates for fatigue cracked specimens, yet calculated leak rates provide a reasonable upper bound to data from the laboratory stress corrosion cracked specimens.

Leak rates at SLB conditions are plotted in Figure 9-3 using data from the Bobbin Voltage ARC database [39]. Data from pulled tubes is plotted along with data from laboratory produced cracks [33]. The scatter in leak rates is relatively large. The SLB leak rate equations lead to calculated leak rates that are near the upper edge of the scatter band of test data. If a reasonable upper bound leak rate is desired, the output of the SLB leak rate equation should be multiplied by a factor of 2 or equivalently by adding 0.060 inches to the leaking 100%TW crack length.

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Figure 9-2
Calculated and Measured Leak Rates for Axial Cracks in Alloy 600 Tubing at Normal Operating Conditions [33]

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Figure 9-3
Calculated and Measured Leak Rates for Axial Cracks in Alloy 600 Tubing at SLB Conditions [33]

9.6 Development of Leakage Paths

The development of 100 %TW degradation can occur over time by the normal processes of degradation growth or mechanical loading of partial depth degradation under normal operating or accident conditions. This can cause failure of the remaining wall thickness leading to a through wall leakage path. This mechanical tearing of material is often referred to as "pop-through". The following three sections provide methods of calculating the occurrence of pop-through for given loading conditions and flaw dimensions for circumferential, axial and volumetric degradation. If pop-through occurs, leak rates are calculated using the approaches of Section 9.4 using the total flaw length as the leaking 100%TW crack size. If through wall penetration does not occur under the loads of interest the leak rate is zero.

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9.6.1 Pop-Through of Circumferential Degradation

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9.6.2 Pop-Through of Axial Degradation

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9.6.3 Pop-Through of Volumetric Degradation

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9.7 Condition Monitoring Evaluation for Leakage Integrity

The leakage integrity requirement for accident induced leakage is that the upper 95th percentile leak rate for each steam generator from all sources shall not exceed the leakage integrity performance criterion. In order to verify leakage integrity during the preceding cycle of operation, each indication for each degradation mechanism shall be evaluated to determine the primary to secondary accident induced leakage rate. The total leak rate for the steam generator shall not exceed the accident induced leakage performance criterion. A total primary-to-secondary accident induced leakage for each degradation mechanism shall account for the inspection sample size. When less than 100% of the tubes are inspected, the CM leakage assessment shall account for possible flaws in locations that are not inspected.

Leakage integrity for cracking type degradation mechanisms is most usefully evaluated in terms of threshold NDE signal amplitudes below which there is no SLB leakage under pressure loading. Values of this type are listed in the Steam Generator In Situ Pressure Test Guidelines [27]. It is possible to use the correlations in [27] for maximum depth versus +Pt™ signal amplitude for circumferential cracks and the pop-through equations of Section 9.6.2 to develop new threshold NDE signal amplitudes for accident leakage that include the effect of bending and axial tensile force loads.

If NDE leakage threshold parameters are not available, lengths and depths inferred from NDE measurements can be used to determine if accident induced leakage would have developed. Lengths and depths leading to 100% through-wall crack tearing under differential pressure, axial tensile force and bending loads can be determined using the equations of Section 9.6.2. The uncertainties appropriate to CM evaluations shall be included for both pop-through analyses and 100% through-wall leaking crack length determinations. Since NDE sizing uncertainties can be substantial, in situ testing to evaluate leakage integrity may be required. Procedures are available listed in the Steam Generator In Situ Pressure Test Guidelines [27] to account for contributing loads by increasing the test pressure.

If accident induced leakage is expected from any detected degradation site, the equations for calculating leak rates as a function of opening area for flow of Section 9.4 may be used. All of these equations are verified versus measured leak rates and measured crack opening areas. Leak rates obtained in this manner can be considered as upper 95th percentile leak rates since they are a good representation of the upper scatter band compared to leak rate measurements from tight stress corrosion cracks. Other approaches to calculating crack opening areas and leak rates are acceptable as long as verification to test results has been performed.

All sources of leakage shall be considered. In fact, plugs and repairs, such as sleeves and re-rolls, may constitute the biggest source of leakage. Qualification of repair products that are of the leak limiting type involves leak rate testing. Appropriate 95th percentile leak rates can be obtained from the qualification reports. Typically, 95th percentile leak rates for each leakage source are directly added to obtain a total leak rate per steam generator. If the distributions of leak rates are known then a Monte Carlo approach can be used to determine the upper 95th percentile total SG leak rate at 50% confidence.

Leakage at normal operating conditions in the previous cycle is monitored during plant operation. Hence, determination of CM operational leakage integrity is performed directly.

9.8 Operational Assessment Evaluation for Leakage Integrity

The potential primary to secondary leakage rate for the most limiting postulated design basis accident, other than a SGTR, shall be assessed. For leakage integrity the fundamental OA requirement is that projected total leak rate from all degradation within a steam generator is less than the leakage integrity performance criteria with 0.95 probability at 50% confidence. Leakage from all sources shall be considered including degraded tubes, plugs, sleeves, rerolls and any other repair process. Upper 95th percentile projected leak rates for each degradation mechanism can be added arithmetically to bound the total SG leak rate. If projected leak rate distributions available for several mechanisms are developed, their contribution to the total projected leak may be combined using probabilistic methods as described in previous sections.

With the bounding worst case degraded tube approach to OA the expectation is that no accident induced leakage will be projected at 0.95 probability with 50% confidence. This is the common circumstance. If leakage is projected for the worst case degraded tube for a given degradation mechanism then the number of projected leakers and the level of leakage is needed for a total leak rate projection. Typically, this then requires use of analytical methods which project the population and severity of leakage sites. Leakage from degradation covered by NRC approved alternate repair criteria requires use of prescribed methodologies that are not presented in this document.

If projected degradation is not calculated to penetrate the tubing wall and projected worst case EOC degradation is not calculated to tear through the remaining wall thickness at accident loading conditions there is no projected OA accident induced leakage. This is a common circumstance in the absence of ARCs and/or when degradation is not within tubesheet crevices where tube burst is prevented. For circumferential cracking and the circumferential extents of volumetric degradation leakage integrity depends on maximum degradation depth and total length. For axial cracking and predominantly axial volumetric degradation under pressure loading flaw shape effects may be included.

Operational assessment projections of leakage at normal operating conditions shall be performed. Low level leakage from leak limiting repair products such as plugs, sleeves and rerolls, and degradation within tubesheet expansion regions shall be included. If alternate repair criteria are implemented, these additional potential leakage sources shall be included. If accident induced leakage is projected for degradation in regions not precluded from burst then potential leakage at normal operation from these sources shall be evaluated.

9.9 Actions upon Failure to Meet Leakage Integrity Performance Criteria

Failure to meet CM leakage integrity requirements is no less serious than failure to meet CM structural integrity requirements. Refer to Section 7.6 for required actions. As with structural integrity, even if CM leakage requirements are met, a comparison of CM leakage results with the projections of the previous OA shall be performed. This comparison shall be completed prior to issuance of the OA since adjustment of input parameters may be required.

10
MAINTENANCE OF SECONDARY SIDE INTEGRITY

10.1 Introduction

The SG program shall include measures to maintain the SG secondary-side integrity as required by NEI 97-06. Monitoring and projecting secondary side steam generator conditions for the purpose of developing a strategy for long-term steam generator operability and performance shall be part of the licensee's steam generator program. This strategy will assist in developing inspection intervals, anticipating future maintenance activities, and planning for contingencies.

The Steam Generator Foreign Object Handbook [43] provides a wide range of information regarding foreign objects such as foreign object control, inspection planning, response, tracking, and trending.

An assessment of secondary side integrity includes reviewing secondary chemistry trends, operational parameters, and inspection data (See Figure 10-1). Examples of potential inputs into the secondary side integrity assessment may include, but are not limited to the following:

1. Steam Generator Design
    a. Materials of Construction
    b. General Design and Configuration
    c. Thermal Hydraulic Information (High Flow Regions, etc.)
2. Secondary Side Chemistry History/Trends
    a. SG Chemistry Excursions (Operating and Shutdown)
    b. Scale and Deposit Removal/Chemical/Profiling Analysis
    c. Corrosion Product Transport and Mass Balance
    d. Hideout Return
3. Secondary Side Maintenance History/Trends
    a. Tubesheet Sludge Deposit Removal
    b. Tube Support Plate / Upperbundle Fouling

c. Foreign Objects

d. Foreign Objects Identified, Removed, and Remaining in the SGs

e. Foreign Objects Identified that Caused Tube Wear

4. Site Specific and Industry Operating Experience (OE)

a. FME or Equipment Degradation Events (SG Foreign Objects Concern)

b. NDE Detectability Issues (Foreign Objects Detection)

c. Secondary Side Visual Inspection Results

d. Secondary Side Component Integrity (Including GL 97-06)

10.2 Purpose

The overall purpose of the secondary side integrity assessment is to provide the SG Engineer with a planning tool for the refueling outage scope to ensure steam generator integrity. This assessment is especially important for SGs that may not perform secondary inspection for multiple operation cycles.

10.3 Secondary Side Assessments and the DA, CM, and OA

Degradation Assessments shall include the secondary side integrity assessment either by directly incorporating the results into the DA or by reference. Refueling Outages that do not include secondary side activities (examples: sludge lancing, FOSAR) shall be evaluated per Section 11.2.4.

Condition Monitoring shall include aspects of the secondary side inspection that affect tube integrity such as secondary side inspections performed, foreign material removed, and foreign material remaining in the steam generators. This information may be included in the CM assessment directly, or with a reference to associated engineering analysis. See Section 7.

Operational Assessments shall include a justification for operating the planned interval between secondary side inspections as well as primary side inspections. See Section 8.

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Figure 10-1
Process of Recording, Monitoring, and Assessing Data

10.4 Secondary Side Cleaning

Steam generator cleaning strategies are either preventive or reactive. Controlling steam generator sludge deposit accumulation through sludge lancing/upper bundle flush every outage supplemented by more aggressive cleaning methodologies at appropriate intervals is a preventive approach. This approach helps prevent potential operational issues that have been associated with sludge accumulation such as tubing corrosion, heat transfer limitation, and water level instability. This approach has lower risk as compared to the reactive approach discussed below.

The reactive approach relies on aggressive cleaning plans for future outages instead of performing cleaning, such as sludge lancing, on a routine basis. The objective of the reactive approach is to restore SG secondary-side cleanliness when long-term planning indicates that operational issues associated with sludge accumulation have the potential to occur. It is essential that steam generator conditions are well understood prior to implementing a reactive approach. The risks with this approach include financial risks associated with a larger number of more aggressive cleaning campaigns and risks with inaccuracies in predicting the onset of operational issues. The other risk involved with this strategy is the potential that foreign objects could remain inside the SGs for a longer period of time, thus potentially increasing the risk of a forced outage due to leakage from foreign object wear.

The secondary side integrity assessment considers the risks and benefits of both strategies and recommends an approach for the next refueling outage(s). While both thermally treated Alloy 600 and 690 tubes are more resistant to alkaline stress corrosion cracking than mill annealed Alloy 600 tubes, consideration should be given to their potential corrosive attack over time. Both thermally treated and mill annealed Alloy 600 tubing are susceptible to acid attack.

As sludge and scale deposits form regions where corrosive impurities can concentrate, care must be taken to minimize their buildup on and around tube surfaces. Due to the different corrosion susceptibilities of the different tubing materials, the more resistant tubing can tolerate sludge deposition to a greater degree. These differences should be considered in formulation of the appropriate cleaning strategy.

There are many secondary side cleaning techniques to be considered when implementing either preventive or reactive cleaning strategies; such as tubesheet sludge lancing, tube bundle flush, scale conditioning agents, deposit minimization treatment, ultrasonic energy cleaning, and chemical cleaning. In addition, improvements in replacement SG designs have incorporated sludge collectors to minimize sludge accumulation at the top of the tubesheet.

10.5 Secondary Side Inspections

For recirculating steam generators, FOSAR shall be performed at a minimum each time sludge lancing is performed. Because of the design of once through steam generators, foreign object intrusion is not expected; therefore, FOSAR shall be performed when loose parts are identified or there is reason to expect that they were introduced into the steam generator secondary side. Secondary side visual examinations should be performed to assist in the verification of tube integrity. The personnel performing secondary side visual inspections and FOSAR activities shall be trained in the use of the equipment and procedures utilized. This training shall include FME control.

Secondary side visual inspections shall include licensee commitments in accordance with NRC GL 97-06, such as visual inspections to detect potential degradation to the wrapper and tube support plates to ensure tube structural integrity is maintained.

An assessment of secondary side tube integrity shall include consideration of operating experience (OE) from all SG models, while paying particular attention to OE from similar SG designs.

An evaluation shall be performed to document the maximum interval between secondary side inspections. This evaluation shall be based on the plant's historical foreign objects, wear indications, maintenance activities, and the planned primary side inspection intervals. The evaluation shall contain the following elements:

- Location and description of historical foreign objects
- Description of those foreign objects with associated wear indications
- High flow, or susceptible areas
- Secondary side inspection limitations
- Trends for foreign object associated wear

When scheduling sludge lancing and FOSAR, the following should be considered:

- Sludge lancing tends to sweep foreign material that is in-bundle toward the annulus, which may allow for easier retrieval.
- Foreign object visibility is often increased when the sludge pile is minimized or removed prior to FOSAR.
- If it is desired to visually confirm a suspected foreign object's position as it relates to a reported wear indication, it may be easier to do so prior to sludge lancing, as sludge lancing has the potential to move the foreign object away from the wear scar.

Several plants have experienced problems with foreign objects after steam generator replacement. During assembly and shipment, SGs are typically kept on their side. When the SGs are set in place during installation, foreign material and debris could fall to the tubesheet and become accessible. Therefore, FOSAR should be performed during the SG replacement outage, after the SGs are installed. In addition, many replacement steam generators have incorporated foreign object strainers, typically as part of the feed ring design, to minimize foreign objects from entering the SG tube bundle region during operation [44]. These strainers should be routinely inspected and considered when performing as assessment of secondary side tube integrity.

Depending on the SG design and foreign object properties (mass, size, etc.), foreign objects entering the secondary side of the SG may locate on the tubesheet within the shell-to-tube bundle annulus region or the blowdown lane (tube lane). SG tubes are typically susceptible to foreign object damage in regions of high secondary feedwater velocity. Therefore the tubes near the shell-to-tube bundle annulus region (the periphery tubes) are typically most susceptible to flow induced foreign object tube wear/damage. When performing FOSAR, the minimum regions to be examined shall include the shell-to-tube bundle annulus region (including periphery tubes) and the tube lane. The scope of the inspection shall be defined in the DA. Visual inspection of the periphery tubes may be achieved by articulating the camera angle to view into the bundle from the annulus region, without inserting the video equipment into the bundle. Visual inspections conducted in this manner provide reasonable assurance that foreign objects with potential to damage tubes located on the secondary face of the tubesheet will be identified, to the extent practical. The above inspection recommendation may not apply to SGs with unique design features such as the main feedwater entering the tube bundle over a baffle plate in the preheater. In such cases, the SG design should be taken into consideration in defining the inspection scope.

All foreign material that has the potential to challenge tube integrity shall be removed from the SGs if reasonably achievable within the limitations of the equipment. Items that are irretrievable and/or could cause damage to tubes by removing them shall be evaluated. Important details to include in the evaluation:

1. An estimation of the material and size of the object (diameter, length, and weight)
2. Location of the object (Tube row/column, Top of Tubesheet (HL/CL), TSP, etc)
3. The estimated axial location of the contact
4. Whether or not the object is firmly lodged or able to move
5. Whether or not tube wear is a result of the object
6. Evaluation of potential wear rate if the object moves and contacts tubes for the planned inspection interval. This evaluation should include conservative assumptions regarding the object's size, the material of the object, tube vibration amplitudes and cross flow fluid velocities.
7. Whether historical NDE (e.g. eddy current data, etc) shows the presence of the object or tube degradation.

When irretrievable foreign material has been identified, it should be inspected in future primary side and secondary side inspections. Irretrievable foreign material that has been determined to have caused tube wear based on NDE, or tube wear is potential based on engineering analysis, shall be inspected in future scheduled SG inspections. Engineering analysis shall determine the inspection interval. Foreign material removed from the steam generators shall also be documented. The type of material entering the SGs and potential for tube damage shall be included in the analysis when determining the interval between primary or secondary side inspections. The EPRI Steam Generator Degradation Database (SGDD) provides a means for documenting and trending foreign material and associated tube damage. It is recommended that (in addition to the SGDD) the licensee maintains and controls documentation related to foreign material tracking.

When potential loose parts (PLPs) are identified during the ECT inspection, they shall be further dispositioned. Options for dispositioning include performing a visual inspection in the area where the PLP was identified by ECT, reviewing historical and current ECT data for PLP signals and wear, bounding the area with a qualified technique, reviewing past visual recordings in the area, and performing an engineering analysis to justify no tube integrity impact without need for search and/or retrieval.

If secondary side inspection confirms the presence of foreign objects, the description and location of the parts shall be documented for consideration during eddy current examination. If primary side eddy current inspections are scheduled, the tubes in the area of the foreign objects should be examined with probes/techniques capable of detecting tube wear. Visual inspections may be considered as an alternative to eddy current inspection, but only if the visual quality and coverage is sufficient to convincingly demonstrate that tube damage is not present in the areas that could have been affected by the part(s). If a part is small enough to enter the tube bundle, visual inspection coverage should include the entire circumference of the tubes in areas potentially affected by the part. Tube integrity shall be evaluated with qualified technique(s) if tube damage is detected (i.e., not superficial surface marks) or considered potential based on visual inspection.

When a foreign object is reported, it is possible that either primary or secondary side inspections are not available. For example, the secondary side integrity assessment could recommend performing FOSAR when primary side inspections and secondary side cleaning are not planned. A similar case is where primary side inspections are performed with no secondary side inspections planned. In either case, planning should include consideration in the event foreign material is identified. When both primary and secondary inspections are performed, these activities should be coordinated to ensure that potential foreign objects identified by eddy current are able to be investigated by the secondary side crew. Similarly, foreign objects identified by the secondary side crew should be communicated to the eddy current leads to evaluate eddy current data for wear, if necessary. Figure 10-2 illustrates recommended logic that should be considered in the event that foreign material is reported by NDE such as ECT and/or SSI.

10.6 Upper Internals Inspections

The inspection frequency of the upper steam drum internals shall be evaluated and documented for plants with recirculating steam generators to verify tube safety functions are not jeopardized by internals degradation. This inspection looks for evidence of corrosion, erosion, chemical deposits, or other conditions which may be present in the upper shell internals. Frequency of these inspections should be commensurate with observed and potential degradation.

Areas of inspection should include:

- Drain pipes and seal buckets
- Instrumentation taps (level transmitters, etc.)
- Demister banks
- Deck plates
- Downcomer barrels
- Wrapper transition to swirl vanes
- Primary separators
- Feedwater ring, feed ring components, and support straps
- Swirl vanes
- Orifice rings
- Applicable welds
- Nozzles

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Figure 10-2
Contingency Planning for Secondary Side Inspection

11
REPORTING

11.1 External Reporting

Required reporting to the NRC is addressed in each licensee's technical specification. This includes reporting failure of CM (See Section 7.6). Deviations taken to this guideline are also required to be reported to the NRC [3]

Required reporting to the industry is addressed in NEI 97-06 and the SGMP Administrative Procedures. It is important the licensees share experiences with the industry in a timely manner through the SGMP and/or the INPO OE process. If a performance criterion is exceeded or if a new industry degradation mechanism is identified, this information should be sent to appropriate SGMP representatives as soon as possible via e-mail so that lessons learned can be disseminated quickly to the industry.

The EPRI SGMP Administrative Procedures include a requirement to identify emergent materials issues with generic significance to the SGMP Program Manager or Integration Committee (IC) Chair as soon as possible. Examples and protocol are included in the Administrative Procedures.

All appropriate tables in the EPRI Steam Generator Degradation Database shall be completed within 120 days after startup.

11.2 Internal Reporting

The reporting discussed in this section is not meant to cover all required internal reporting or documentation. This section is concerned with required reporting for integrity assessments. Refer to other EPRI Guidelines for additional internal reporting requirements.

11.2.1 The Degradation Assessment Report

The Degradation Assessment Report is the documentation of the DA process and shall be completed prior to each scheduled steam generator inspection. Aspects of the DA Report that should be drafted well in advance of the final report are scope of the inspection, techniques that will be used, and number of tubes that are predicted to be plugged or repaired. This will support outage planning and scheduling and purchasing of probes and plugs or sleeves. Elements of the DA process which are required to be documented in the DA Report are specified in Section 6. The Degradation Assessment Report shall be developed by or reviewed and approved by a licensee representative knowledgeable of the DA and CM/OA process and shall be a retrievable licensee document.

11.2.2 The Condition Monitoring Report

The Condition Monitoring Report is the documentation of the CM process and shall be completed prior to MODE 4 after a SG inspection. The CM process should be an ongoing process throughout the SG inspection. It is recommended that screening of indications for additional NDE or in situ pressure testing be completed prior to plugging or repairs, but shall be completed prior to MODE 4 after a SG inspection. The CM report shall be developed by or reviewed and approved by a licensee representative knowledgeable of the DA and CM/OA process and shall be a retrievable licensee document.

11.2.3 The Operational Assessment Report

When SGs pass CM, an OA shall be completed for the next inspection interval within 90 days after MODE 4. If completion of this assessment is not possible due to the complexity of the analysis within the 90-day period, a documented evaluation shall be completed that provides reasonable assurance that the performance criteria will not be exceeded between the 90 days and the date of the OA report. The 90 day time period is acceptable because of successful completion of CM and the application of conservative repair limits.

When SGs fail CM or the results are indeterminate, a documented evaluation shall be completed prior to MODE 4 that provides reasonable assurance that the performance criteria will not be exceeded between MODE 4 and the date of the OA report. The OA report shall be completed for the next inspection interval within 90 days after MODE 4. If completion of this assessment is not possible due to the complexity of the analysis within the 90-day period, a documented evaluation shall be completed that provides reasonable assurance that the performance criteria will not be exceeded prior to the date of the OA report. Additional reporting requirements are necessary, in accordance with NEI 97-06 [2] and the plant's Technical Specifications.

The Operational Assessment Report shall be performed by or reviewed and approved by a licensee representative knowledgeable of the CM/OA process and shall be a retrievable licensee document.

11.2.4 Review of SG Integrity Assessment Documents Prior to a Refueling Outage without SG Inspections

A review of the SG integrity assessment documents that justify the planned inspection interval shall be performed prior to each refueling outage when SG primary and/or secondary-side inspections are not scheduled, to validate the inspection interval. The review shall include consideration of the DA and OA that justify the planned inspection interval, and events subsequent to the DA and OA such as: industry operational experience, chemistry excursions and transients, and plant operating transients that might impact SG tube integrity since the last inspection and/or review. This review should be performed in a timely manner in order to minimize outage impact (scope/budget/planning - critical path), should a change in the planned inspection interval be identified from the review.

The review of the SG integrity assessment documents that justify the planned inspection interval shall be performed by or reviewed and approved by a licensee representative knowledgeable of the DA and CM/OA process and shall be a retrievable licensee document.

12
REQUIREMENTS AND RECOMMENDATIONS

In accordance with NEI 03-08, this section clearly identifies mandatory and shall (or needed) requirements and recommended (or best practice) elements of this guideline document.

12.1 Introduction

12.2 Tube Integrity Criteria

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12.3 Tube Integrity Assessment Limits

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12.4 NDE Measurement Uncertainties

Proprietary Information

12.5 Degradation Growth Rates

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12.6 Degradation Assessment

Deleted- EPRI Proprietary Information

Deleted - EPRI Proprietary Information

12.7 Condition Monitoring

Deleted EPRI Proprietary Information

12.8 Operational Assessment

Deleted - EPRI Proprietary Information

12.9 Primary-to-Secondary Leakage Assessment

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Deleted - EPRI Proprietary Information

Deleted- EPRI Proprietary Information

12.10 Maintenance of SG Secondary Side Integrity

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Deleted - EPRI Proprietary Information

12.11 Reporting

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13
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A
APPENDIX A: GLOSSARY

Accident Induced Leakage

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

Active Degradation[1]

Term used in the Steam Generator Technical Specifications synonymous with the term existing degradation used in this guideline.

Alternate Repair Criteria (ARC)

Alternate 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.)

Beginning of Cycle (BOC)

The start of an inspection interval.

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.

Characteristic Shape Factor

The ratio of maximum degradation depth to structural average degradation depth.

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[1]The SGMP PWR Steam Generator Examination Guidelines [1] defines degradation as a reportable indication 20% TW or greater or 50% of the repair limit for length-based or voltage-based criteria. Reference 1 uses the term as it applies to inspection scope and expansion. The definition in the Steam Generator Integrity Assessment Guidelines is different in that it uses the term as it applies to integrity assessments. All degradation, no matter what the size, is assessed for tube integrity.

Collapse

For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero.

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 inspection.

Condition Monitoring Limit

The limiting magnitude of a degradation measurement parameter, such that the tube section satisfies the applicable performance criterion with 0.95 probability at 50% confidence.

Defect

A condition that requires repair or plugging of a tube.

Degradation[1]

Tube wall loss (crack-like or volumetric).

Degradation[1]-Specific Repair Criteria

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

Degradation[1] Mechanism

Refers to a process causing degradation, such as wear, pitting, thinning, ODSCC, PWSCC, etc.

Discrete Sample

An area of steam generator tubing which, on the basis of inspection results, engineering evaluation and related experience, is defined by the type of degradation, the cause of degradation, and the boundary of degradation. See SGMP PWR Examination Guidelines [1].

EFPM

Effective full power months of operation.

EFPY

Effective full power years of operation.

End of Cycle (EOC)

Synonymous with end of inspection interval for the purposes of this document.

EOC Structural Limit

The predicted actual value of the specified limiting structural parameters at the EOC (e.g., actual length and depth).

Existing Degradation[1]

Indications of degradation previously and/or currently observed in a SG.

Extreme Values

The highest or lowest (whichever is applicable) in a group of values. For example: the extreme burst pressure would be the lowest calculated burst pressure.

Flaw

Degradation identified by visual examination, tube pull, or actual leakage. NDE indications are evaluated as flaws in integrity assessments.

Indication

The response or evidence from the application of a nondestructive examination. NDE indications are evaluated as flaws in integrity assessments.

Inspection Interval (II)

Interval between inspections of the SG tubing measured in EFPM.

Limiting Accident

For structural analyses: An accident that results in the largest differential pressure across the steam generator tubes, normally a main steam line or main feedwater line break.

For leakage analyses a review of the plant's design basis accidents is required to determine the limiting accident in terms of leakage. Some design basis accidents assume primary-to-secondary leakage. For plants that assume the same leak rate in each design basis accident (DBA), one AILPC limit shall be established using the assumed leak rate combined with the highest accident loading condition on the tubing. For plants that do not assume the same leak rate in each DBA, utilities shall use one of the two options below:

1. The loading condition on the tubing for each accident can be evaluated along with the assumed leak rate to define an AILPC limit for each accident, or
2. The lowest leak rate assumption can be combined with the highest tube loading condition to define one bounding AILPC limit.

Lower Tolerance Limit

A value or event that would be expected to occur with a probability of P and a confidence level of C. For example, a lower 95%/50% lower tolerance limit for material strength means a 50% confidence that ≥ 95% of the population of yield strengths are greater than the specified lower tolerance limit. Note that any value of P and/or C can be chosen to define a lower tolerance limit.

NDE Measurement Parameter

A variable used to measure degradation extent, such as voltage, crack length, depth etc.

NDE System

Inspection techniques, personnel, and procedures used to perform non destructive examination.

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 their effects on differential pressure included if significant.

Normal Operating Pressure Differential

The pressure differential across the tube wall during normal steady state full power operation.

Operational Assessment

Forward looking evaluation of the steam generator tube conditions that is used to ensure that the structural integrity and accident leakage performance criteria will not be exceeded during the next inspection interval. The operational assessment needs to consider factors such as NDE uncertainty, indication growth, and degradation-specific repair limits.

Operational Assessment Limit

The value of a degradation parameter such that a tube with greater degradation at the beginning of an operating period would not meet the performance criteria at the end of the operating period with 0.95 probability at 50% confidence.

Percent Degraded Area

For a circumferential crack, the percentage of cross-sectional degraded area relative to the total cross-sectional area of the tube.

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.

Potential Degradation[1]

Mechanical or corrosive process that has not been discovered in prior inspections in the SGs but are judged to have a potential to occur in the current inspection period based on industry experience and/or laboratory data. All tubing locations associated with Potential Degradation Mechanisms must be inspected during each technical specification inspection period.

Proactive Inspections

Supplemental inspections that are not required which are performed for informational purposes and are not expected to result in findings. These exams differ from those exams performed for Existing or Potential Degradation in that they do not fall within the periodicity sample requirements.

Probabilistic Approach

An approach that uses probabilistic simulations, e.g., Monte Carlo simulations, to determine appropriate limits.

Probability of Burst (POB)

The relative frequency of gross structural failure of a steam generator tube under a postulated loading condition.

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 according to Generic Letter 95-05).

Repair Limit

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

Repair on Detection

No detected degradation is intentionally left in service.

Repair on NDE Sizing

Detected degradation is intentionally left in service if the size determined by NDE is below a certain level.

Root Cause Evaluation

An analysis that is performed to determine the fundamental cause(s) of a problem or issue. The analysis' level of rigor is based on the significance of the issue and is per the licensee's corrective action procedures.

Secondary Stress

Secondary stress is the stress component developed by the constraint of adjacent material or by self-constraint of the structure. Secondary stresses, by definition, are self-limiting in nature. The basic characteristic of a secondary stress is that local yielding or deformation will reduce (or eliminate) the load and resulting stress. In addition, failure from one application of a secondary stress is not expected to occur.

Significant Loads

An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established.

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

Unless otherwise defined by Technical Specifications (i.e., Alternate Repair Criteria), 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.

Structural Average Depth and Structural Length

The structural average depth and structural length of axially oriented degradation are the combination of these variables consistent with segments of the depth profile that lead to a minimum estimate of burst pressure [5]. This definition is applicable to structural depth axial cracks and volumetric degradation that is less than 135 degrees in circumferential extent.

Structural Limit

The limiting degradation parameter that will satisfy the structural integrity performance criterion based on nominal tube material properties and dimensions, and the nominal predictions of a regression curve.

B

APPENDIX B: LIST OF ABBREVIATIONS AND ACRONYMS

AILPC	Accident-Induced Leakage Performance Criterion
ARC	Alternate Repair Criteria
ASL	Axial Secondary Loads
ASME	American Society of Mechanical Engineers
AVB	Anti-Vibration Bar
AVT	All Volatile Treatment
BOC	Beginning of Cycle
BP	Burst Pressure (some calculations refer to burst pressure as P_B or P_{Burst})
B&W	Babcock and Wilcox
CAF	Corrosion-Assisted Fatigue (High Cycle Fatigue)
CDF	Cumulative Distribution Function
CE	Combustion Engineering
CEOG	Combustion Engineering Owners' Group
CFR	Code of Federal Regulations
CLT	Cold Leg Thinning
CM	Condition Monitoring
CMTR	Certified Mill Test Report
DA	Degradation Assessment
ECT	Eddy Current Testing
EDF	Empirical Distribution Function
EFPY	Equivalent Full-Power Years
EOC	End of Cycle
EPRI	Electric Power Research Institute
ETSS	Examination Technique Specification Sheet
FDA	Fractional Degraded Area
FLB	Feed Line Break
FME	Foreign Material Exclusion
FOSAR	Foreign Object Search and Retrieval
FS	Free Span

FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GLM	Generalized Linear Modeling
GPD	Gallons Per Day
GR	Growth Rate
IAGL	Integrity Assessment Guidelines
ICC	Intergranular Cellular Corrosion
ID	Inside Diameter
IGA	Intergranular Attack
IDIGA	Inside Diameter Intergranular Attack
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute for Nuclear Power Operations
ISI	In-Service Inspection
LAPD	Limiting Accident Pressure Differential
LCO	Limiting Condition Operation
LOCA	Loss of Coolant Accident
LR	Leak Rate
LTL	Lower Tolerance Limit
MAPOD	Model-Assisted Probability of Detection
MD	Maximum Depth
MSLB	Main Steam Line Break
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NMP	NDE Measurement Parameter
NOP	Normal Operating Pressure
NOPD	Normal Operating Pressure Differential
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OA	Operational Assessment
OD	Outside Diameter
ODSCC	Outside Diameter Stress Corrosion Cracking
OE	Operating Experience
OTSG	Once Through Steam Generator
PDA	Percent Degraded Area
Pdf	Probability Density Function
PL	Primary Load
POB	Probability of Burst

POD	Probability of Detection
POL	Probability of Leak
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RCPB	Reactor Coolant Pressure Boundary
RFC	Retirement for Cause
RG	Regulatory Guide
RPC	Rotating Pancake Coil
RSG	Recirculating Steam Generators
SF	Safety Factor
SG	Steam Generator
SGDD	Steam Generator Degradation Database
SGMP	Steam Generator Management Program
SGPB	Steam Generator Pressure Boundary
SGTR	Steam Generator Tube Rupture
SIPC	Structural Integrity Performance Criterion
SL	Structural Limit
SLB	Steam Line Break
SR	Stability Ratio
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
STP	Standard Temperature and Pressure (60°F, 760 mm Hg)
S/N	Signal-to-Noise Ratio
TRM	Technical Requirements Manual
TS	Tubesheet
TSP	Tube Support Plate
TSPC	Tube Support Plate Center
TSPE	Tube Support Plate Edge
TW	Through Wall
UTS	Upper Tubesheet (in OTSGs)
W	Westinghouse

C

APPENDIX C: INDUSTRY TECHNICAL BASES FOR STRUCTURAL INTEGRITY ASSESSMENT

C.1 Introduction

General Design Criteria (GDC) 1, 2, 4, 14, 30, 31 and 32 of 10 CFR Part 50, Appendix C, define requirements for the reactor coolant pressure boundary (RCPB) with respect to structural and leakage integrity [C1]. 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 able to maintain reactor coolant inventory and pressure. The structural integrity performance criterion (SIPC) from Nuclear Energy Institute (NEI) 97-06 [C2] was developed to provide reasonable assurance that a steam generator tube will not burst during normal or postulated accident conditions.

The NEI Steam Generator Task Force and the EPRI SGMP endeavored to define integrity assessment requirements that facilitate the CM and OA process. This effort resulted in the publication of three (3) industry position papers for the definition of tube burst/collapse, loading definitions, and design safety margins supporting the SIPC. These white papers provide a historical and technical compendium of information defining the basis for these integrity assessment criteria. The purpose of Appendix C is to document the complete technical basis from these white papers in one location within the Steam Generator Integrity Assessment Guidelines.

It is important to note that the safety factors appearing in the SIPC defined in Section 2 for integrity assessment apply to specific pressure conditions and should not be applied to loads that are not identified in the industry definition. However, Section 2 notes that meeting the SIPC ensures meeting the requirement that tubes are not stressed beyond the yield strength of the tubing for the expected tube loadings. Thus, the integrity evaluation would verify that the primary membrane stress does not exceed the yield strength for the full range of normal operating conditions as described in the performance criteria. In addition, other (non-pressure) loads contributing to combined primary plus secondary stress, as identified in the design and licensing basis, are evaluated so as to ensure that these loads do not significantly contribute to burst for the postulated accidents.

As with the industry position papers, Appendix C presents historical documentation and technical information demonstrating that there has been a consistent industry and regulatory approach for this definition of structural integrity and the associated margins against tube burst. In this current effort, the SGTF has endeavored to provide a link between the revised loading definition and ASME Code definitions and requirements, historical regulatory requirements, and industry protocols previously approved by the US Nuclear Regulatory Commission (NRC). As industry protocols previously approved by the US Nuclear Regulatory Commission (NRC). As such, information provided in Appendix C demonstrates that the recommended criterion is consistent with past practices and meets the intent of the original design basis for steam generator tube integrity. Meeting this definition will ensure the nuclear safety of in-service steam generator tubing and associated repairs, and minimizes the potential for adversely impacting the design function of the steam generator.

C.2 Definition of Burst

C.2.1 Burst Condition

The definition of tube burst used in tube integrity assessments for CM and OA is given below:

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The condition of tube burst/collapse is an important element of the structural integrity performance criterion (SIPC) discussed further below.

C.2.2 Technical Discussion

A burst 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 [C3], and compliance with the historical guidance of the Draft Regulatory Guide 1.121 [C4]. The definition of burst according to 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 Boiler and Pressure Vessel Code” [C5]. Burst, or gross failure, according to the Code would be interpreted as a catastrophic failure of the pressure boundary.

The definition not intended 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-inch 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 radial direction may still be demonstrated. Similar conditions have been observed for deep wear indications.

Additional information in Reference C7 further supports the proposed definition of burst. It is noted that if tube failure, i.e., burst, is defined “as plastic deformation of the crack to the extent that the sides of the crack open to a non-parallel elliptical configuration, the tubing can sustain added internal pressure beyond those values before reaching a condition of gross failure.”

C.2.3 Application - Condition Monitoring

Verification of structural integrity during CM may be accomplished via in situ testing. The EPRI *Steam Generator In Situ Pressure Test Guidelines* [C8] provide industry guidance regarding the evaluation of field test results. The industry guidance for the conduct and application of laboratory burst tests can be found in the Guidelines for Steam Generator Tube Section Removal, Test and Examination, and the EPRI Leak Rate Testing and Burst Testing Guidelines [C9-C10].

C.3 Deterministic Structural Performance Criterion Pressure Loading Definition

C.3.1 Background

In response to NRC and industry comments to NEI Generic License Change Package (GLCP), the NEI SGTF, with support from EPRI SGMP, reevaluated the SIPC in NEI 97-06 with regard to required safety margins. Based on the results of the evaluation, the SGTF developed the technical basis supporting a revised definition of the deterministic structural integrity performance criterion. This definition addresses all comments and provides a more generic and universal criterion that is considered applicable to all steam generator designs. The new definition is included in NEI 97-06, Revision 2 [C2], applicable EPRI Guidelines, and the GLCP.

C.3.2 Statement of Structural Performance

The structural integrity performance criterion for demonstrating the structural integrity of steam generator tubing is stated below:

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The SIPC is based on loading definitions and an evaluation framework consistent with the ASME Code and past draft regulatory guidelines. The logic of the SIPC illustrating the key assessments for integrity is given in Figure C-1. This figure shows the general approach in establishing structural limits for degraded tubing. Discussion of the determination of allowable structural limits is given in Section C.7.

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Deleted EPRI Proprietary Information

Figure C-1
SIPC Implementation Logic

C.3.3 Definitions

The definition of tube burst is given in Section C.2.1. Definitions for the other terminology within the SIPC are given below:

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A summary discussion of the important elements of the performance criterion is given in Section C.3.4. Detailed information regarding the basis for key components of the criterion statement, including required safety factors, is provided in Sections C.4 through C.6.

C.3.4 Technical Discussion

The imposition of a safety factor of 3.0 against burst under the normal steady state full power primary-to-secondary pressure differential (3NOPD) is to ensure that the overall tube integrity is maintained for all normal operating and upset conditions and can be verified through condition monitoring. As indicated in the industry position paper, the use of normal steady state full power operating pressure differential has been consistently utilized as the basis for compliance with the 3NOPD criterion. The Technical Specification plugging limits for the plants surveyed in support of Appendix C were all based on full power differential pressure. Likewise, the surveyed design reports for minimum wall calculations for tubing and sleeves and Draft Regulatory Guide 1.121 [C4] compliance calculations were based on normal full power differential pressure. The revised performance criterion makes no changes to this part of the definition. Instead, this effort focused on the correct application of the 1.4 safety factor for accident conditions and the necessary adjustment for loading conditions that may contribute to burst.

With respect to demonstrating structural integrity for faulted (e.g., accident) conditions, the SGTF determined that historical precedence dictated that the safety factor of 1.4 on primary membrane load be applied only to those conditions specified as Service Level D in the design specification. The inclusion of a safety factor of 1.4 on Service Level D events ensures that adequate margins against tube burst will be maintained for the applicable postulated accident events. The technical bases for these margins have their origins within Section III design rules of the ASME Code. Historically, faulted (Level D) conditions have been selected for evaluation of limiting design basis accidents by regulatory precedence. These postulated accident events generally impose the most limiting conditions for primary-to-secondary differential tube pressures and therefore should be bounding for the design basis accidents, including Level C events [C12]. However, to capture all the design basis transients, the criterion was written to establish the structural limits for accidents that also include pressure differentials from Level C events. This will assure that the largest differential pressure is used in defining the limits based on all design basis accidents specified for the plant.

The revised definition also includes an assessment of loading conditions defined in the design and license basis that could potentially contribute to reducing the tube burst pressure. The assessment of these additional conditions is to assure that the associated loading conditions that can significantly contribute to tube burst are evaluated. This general assessment is needed so that the structural integrity performance objectives are adequately satisfied, given that there are design-specific as well as plant-specific situations that cannot be generically addressed. For example, such loads would include other primary loads from dynamic events, and axial thermal loads resulting from differential temperatures created during transients in once-through steam generator (OTSG) designs, or similar loads associated with locked tube supports which could be postulated to develop in recirculating steam generator (RSG) designs. When it is determined that these loads significantly affect tube burst conditions, a safety factor of 1.2 is applied to combined primary load sources, and a factor of 1.0 is applied to axial secondary loads.

Further discussion of the technical basis for each of the specified safety margins is given in Section C.4 and C.5. Discussion of the basis and assessment of additional contributing loads is given in Sections C.6 and C.7. A technical discussion on the treatment of axial thermal loads is given in Section C.6.4.

C.3.5 Limits on Yield Strength

Historical regulatory precedence dictates that tube integrity be maintained for the full range of reactor operation including startup, operation in the power range, hot standby and cooldown, and all anticipated transients that are included in the design specification. Hence, a key aspect of tube integrity as stated in Draft Regulatory Guide 1.121 [C4], NEI 97-06 [C2], EPRI Integrity Assessment Guidelines, and the GLCP is to ensure that degraded steam generator tubes are not stressed beyond the elastic range of the tube material for primary membrane loads for normal operating conditions (i.e., gross yielding of the tube is prevented during normal operation). In this context, tube integrity for all expected operating scenarios (i.e., the full range of reactor operation including startup, operation in the power range, hot standby and cooldown, and all anticipated transients) is demonstrated if the associated primary membrane stresses are below the yield strength of the tube material for Service Level A and B conditions. As it is not practical to verify a yield strength criterion in a field application (e.g., in situ pressure testing), industry has used the applied safety factors on pressure loading and a specified definition of burst to ensure the yield strength criterion is also satisfied. Meeting the structural requirements of the SIPC will limit the primary membrane stress in the degraded region to meet the intent of Draft Regulatory Guide 1.121. Therefore, meeting the limiting conditions of the SIPC ensures meeting the requirement that tubes are not stressed beyond the yield strength of the tubing. No further evaluation beyond the assessment for margins defined in the SIPC is necessary to prove this criterion on gross tube yield is met.

C.4 ASME Code Review

C.4.1 Minimum Wall Requirements

A review of multiple industry documents, including NSSS design reports, sleeving topical reports, and regulatory submittals, indicates that all NSSS designers and multiple licensees have adopted the same ASME Section III Code equation, from paragraph NB-3324.1 to define minimum allowable tube wall thickness. The equation in NB-3324.1 is used to calculate a tentative wall thickness for use in component design. There are no requirements to account for inservice degradation. The equation as specified in the ASME Code is as follows:

$$t = \frac{PR}{S_m - 0.5P}$$ **Equation C-1**

where:

- t = tube wall thickness
- P = design pressure
- R = tube inside radius
- S_m = design stress intensity

In a typical design scenario, a tube wall thickness is selected based on the results of this calculation. The proposed tube size is then assessed against all other ASME required design analyses including but not limited to bending loads, combined primary plus secondary stresses, seismic loads and fatigue loads.

In establishing plugging limits, which according to Draft Regulatory Guide 1.121 [C4] should account for inservice degradation growth and NDE uncertainty, licensees and NSSS vendors were required to define a minimum acceptable wall thickness. A survey of industry documents indicates a consistent application of the definitions contained in NB-3324.1. A typical interpretation of NB-3324.1 is as follows:

$$t = \frac{\Delta P R}{P_m - 0.5\Delta P}$$ **Equation C-2**

where:

- ΔP = primary-to-secondary differential pressure
- P_m = primary stress limit, i.e.,
- P_m normal full power operation $\leq S_u/3$
- P_m normal and upset (Level A and B) $\leq S_y$
- P_m faulted (Level D) $\leq 0.7S_u$

The above primary stress limits are consistent with the ASME Code design requirements, which are explicitly maintained in the assessment of tube integrity. Therefore, the Code design margins of 3.0 and 1.4 (i.e., 1/0.7) form the basis for the structural integrity performance criterion for differential tube pressure as stated in Section C.3.2.

C.4.2 Primary Loads from Accident Events

C.4.2.1 ASME Section III Appendix F Considerations

The basis for the structural integrity performance criterion for primary membrane stress for the design basis accidents is NB-3324.1 of ASME Section III, as previously discussed. This includes application of Appendix F of ASME Section III. Appendix F defines a margin on primary membrane stress of 1.4 derived from the allowable Level D limit of 0.7 S_u. Therefore, imposing a safety factor of 1.4 on primary membrane load due to differential pressure will preserve the elastic stress limits provided by the code. It has been shown that primary membrane loads other than differential pressure are negligibly small [C4].

When an analysis is performed to ASME Section III NB-3200, the evaluation of accident loads may be performed in accordance with the rules of Appendix F. For the case of combined primary membrane plus bending, the Appendix F collapse method can used to define acceptance for tube integrity. Acceptance criteria for elastic system analysis are covered in Article F-1331.1(c)(2) and Article F-1341.3 covers plastic system analysis. Both methods and acceptance limits are essentially the same. Appendix F Subsections F-1331.1(c)2 and F-1341.3 define the allowable stress for the analysis of collapse as 90% of the calculated collapse load where the critical net-section stress is not to exceed the lesser of $2.3S_m$ or $0.7S_u$. Therefore, an analysis could use this allowable and meet the requirements of the ASME Code. The factor of safety implied from this requirement is 1.11. It is apparent that the plastic collapse analysis has different inherent design margins than the one performed using the elastic allowable stress limits.

The Code collapse load is the maximum load that the tube can withstand before deformation of the tube will increase without limit. This definition for maximum load is consistent with the industry definition of the tube burst condition. When the Appendix F collapse method is applied, a safety factor of 1.2 can be shown to be a conservative factor to be applied to combined primary loads in an analysis for tube burst. This is outlined below.

Determination of the 1.2 safety factor (SF) is derived from an equivalency comparison between the load that would be permitted by the Appendix F Code collapse method and the collapse load determined by standard industry methods for tube integrity with circumferential degradation:

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Table C-1
Alloy 600 Typical Properties – Mean Values

C.4.2.2 ASME Section XI Pipe Flaw Assessments

The design of steam generator tubing is dictated by ASME Section III design rules for vessels covered under NB-3300. However, a general review of the pipe design rules under NB-3650, and the flaw evaluation procedures for flawed piping in ASME Section XI [C13] in IWB-3640, provide some insight on the technical basis for which safety factors have been developed within ASME Section XI for situations where combined membrane plus bending loads are addressed for plastic collapse. The derivation of the explicit "combined" safety factors for Class 1 piping is based on the logic discussed in [C14]. These safety factors have been recently refined in [C15], where more care was taken to make equivalent the safety factors used in Section XI with the design margins in Section III, in this case with the piping design rules of NB-3650. In both situations, the linkage to the Section III Code design bases was made by defining the safety factor as

$$SF = \frac{Section\,XI\ Collapse\ Stress}{Section\,III\ Alowable\ Stress}$$ **Equation C-5**

From Equation. C-5, and the design conditions from NB-3641 and NB-3652, the analysis safety factors for pure membrane and pure bending loads are,

$$SF_m = \frac{\sigma_{Flow}}{C_p\,S_m}$$

$$SF_b = \frac{4}{\pi}\,\frac{\sigma_{Flow}}{f\,S_m}$$

where C_p is the permissible pressure factor and f is the primary stress limit factor given by Code Equation 9 in NB-3652, which are both dependent on Service Level (i.e., C_p = 1.0, 1.1, 1.5, and 2.0; and f = 1.5, 1.8, 2.25 and 3.0,for Service Levels A, B, C, and D).

The safety factors used in Class 1 piping in ASME Section XI since the mid 1980s were the result of averaging safety factors for membrane and bending values for Service Levels A and D. The same factors were then applied to Levels B and C, and the corresponding safety factors for evaluation became 2.77 for Levels A and B and 1.39 for Levels C and D. Since we are concerned with the justification of appropriate factors for accident events, the original derivation of the safety factor equal to 1.39 for piping is given below:

$$SF = \frac{1}{2}[SF_b + SF_m] = \frac{\sigma_{Flow}}{2S_m}\left[\frac{4}{\pi k} + \frac{1}{C_p}\right]$$ **Equation C-6**

With $\sigma_{Flow} = 3S_m$ for austenitic stainless steels, f = 3.0, and $C_p = 2$, for Level C/D conditions,

$$SF = \frac{3}{2}\left[\frac{4}{3\pi} + \frac{1}{2}\right] = 1.387$$

The above calculation is the basis for the 1.39 factor for evaluating flawed austenitic piping up to the 2001 Edition of the ASME Code.

The definition of flow stress as $3S_m$ for austenitic piping was based on lower bound estimates of flow stress from 26 samples of Type 304/316 wrought pipe material and 11 samples of Type 308 weld metal, all tested at 550°F [C14]. This definition is unique to austenitic stainless steels and determined to be an overestimate for ferritic steels and nickel-based alloys. In this manner, the safety factors for piping products fabricated from Alloy 600 and ferritic steels would be bounded.

Subsequently, a revised definition for σ_{Flow} has been developed based on the average of S_y plus S_u which permits the use of physically-based properties relevant to the specific pipe alloy. This definition of σ_{Flow} is currently in use in Section XI and provides a better measure of the onset of plastic collapse. This definition has been validated with full-scale pipe tests involving part-through-wall circumferential cracks. The details of the use of σ_{Flow} in the definition of safety factors are discussed in more detail in [C15].

With the above refinements on flow stress, one can now evaluate what safety factor would result for Alloy 600/690 material using Equation. C-6. Using the properties for Alloy 600 at 600°F from above as the bounding case:

$$\sigma_{Flow} = 0.5(27.9 + 80) = 53.95\,ksi$$

$$SF = \frac{53.95}{(2)(23.3)}\left[\frac{4}{3\pi} + \frac{1}{2}\right] = 1.070$$

This safety factor is less than the 1.20 value developed from Section III vessel requirements.

The above computed safety factor assumes that $3S_m$ will be controlling in the Code Equation 9 design limit for Level D. The recent changes to the pipe flaw evaluation procedures consider both $3S_m$ and $2S_y$; whichever is less, as the allowable stress. With $2S_y$ as the Code Equation 9 allowable stress, the safety factor relationship becomes

$$SF = \frac{1}{2}[SF_b + SF_m] = \frac{\sigma_{Flow}}{2}\left[\frac{4}{2S_y\,\pi} + \frac{1}{C_p\,S_m}\right] \qquad \textbf{Equation C-7}$$

For Alloy 690 at 600°F as the bounding case,

$$SF = \frac{53.8}{2}\left[\frac{4}{(2)(27.6)\pi} + \frac{1}{(2)(23.3)}\right] = 1.198$$

This computation for safety factor provides a piping design-based value that is consistent to value derived from ASME Section III Appendix F plastic collapse rules.

C.4.3 Secondary Loads from Accident Events

C.4.3.1 Definition of Secondary Loads

Secondary loads produce secondary stresses which have a specific definition and hence, analytical treatment within ASME Section III. Secondary stress is a normal or a shear stress developed by the constraint of adjacent material or by self-constraint of the structure. The basic characteristic of a secondary stress is that it is self-limiting. Local yielding and minor distortions can satisfy the conditions that cause the stress to occur and failure from one load application of the stress is not to be expected.

Thermal loads produce thermal stresses, which fulfill the definition of secondary stresses for Class 1 design. Thermal stresses are self-balancing stress produced by a non-uniform distribution of temperature or by differing thermal coefficients of expansion. Thermal stresses are developed in a solid body whenever a volume of material is prevented from assuming the size and shape that it normally should under a change in temperature. For the purpose of establishing allowable stresses, two types of thermal stress are recognized in Class 1 design, depending on the volume or area of material in which the distortion takes place, as described in (a) and (b) below.

a) General thermal stress is associated with distortion of the structure in which it occurs. If a stress of this type, neglecting stress concentrations, exceeds twice the yield strength of the material, the elastic analysis may be invalid and successive thermal cycles may produce incremental distortion. Therefore this type is classified as secondary stress.

b) Local thermal stress is associated with almost complete suppression of the differential expansion and thus produced no significant distortion. Such stresses are considered only from a fatigue standpoint and are therefore classified as local stresses in Table NB-3217-1. When evaluating local thermal stresses, the procedures of NB-3227.6(b) are be used.

By the above definitions, thermal loads are classified as secondary loads for Level A and B conditions. Constraint on high local strains is imposed through the fatigue design rules, which are crafted to prevent plastic ratcheting. These requirements are to guard against failure in the low-cycle fatigue regime, which would include failure by a single-load application.

C.4.3.2 Code Practice

The design of steam generator tubing is covered by ASME Section III design rules for vessels covered under NB-3300. Secondary loads are not required in Code evaluations of Level C and D stress limits under NB-3200, since they are self-equilibrating and do not significantly affect maximum load capacity once yielding has occurred. However, in tube integrity evaluations to Draft Regulatory Guide 1.121 and in tube burst assessments, axial thermal loads have been historically considered because they may impose large axial strains.

The axial load in a once-through steam generator, which is created primarily by the difference in temperature between the tube and shell, is considered a secondary loading per ASME Code definition. By this definition, secondary load (stress) is developed by the constraint of adjacent material or by self-constraint of the structure. The basic characteristic of a secondary stress is that local yielding or deformation will reduce (or eliminate) the load and resulting stress. The thermal load is self-limiting and will not cause failure of the tube or its repair hardware under single load application. Because of the displacement-controlled nature of these loads, the Code allowables for secondary stresses do not contain a safety factor (i.e., safety factor equal to 1.0).

This design philosophy is also carried into the flaw/toughness evaluation procedures of both ASME Section III and Section XI. For example, the determination of the pressure-temperature heatup/cooldown curves for the reactor coolant system (RCS) is based on a fracture mechanics assessment method where primary stresses are multiplied by a safety factor of 2.0, and secondary stresses are included with a safety factor of 1.0. Similarly for other flaw evaluations involving austenitic materials, thermal expansion stresses are considered with a safety factor of 1.0. ASME Section XI RCS evaluations for pressure boundary integrity, including evaluations of service-induced degradation, consider secondary stresses as being less severe than primary stresses, and generally include them without imposing a safety factor. Therefore, it can be appropriate to consider contributing axial thermal loads with a safety factor of 1.0 as stated in the structural performance criterion. This practice is consistent with the ASME Code.

C.4.4 Summary of Code Considerations

The applied safety factors specified in the SIPC have their bases formulated from design rules for Class 1 vessels in accordance with ASME Code Section III, Subsection NB-3300. This includes preserving a factor of 3.0 against tube burst under normal operating steady-state operation, an assessment to ensure that degraded steam generator tubes not be stressed beyond the elastic range of the tube material for all normal and upset primary membrane loads, as well as appropriate factors for design basis accidents. The safety factors for accident loads include maintaining a safety factor of 1.4 on primary membrane load resulting from differential pressure, and a combined safety factor of 1.2 on primary loads (membrane plus bending) and 1.0 on axial secondary loads. These safety factors on accident loads are consistent with ASME Section XI flaw evaluation acceptance criteria for flawed piping when reformulated for Alloy 600/690 mechanical properties. Historically, thermal loads have been considered as secondary in both Section III design and Section XI flaw evaluations.

C.5 Historical Perspective

C.5.1 Regulatory Perspective

Nuclear Energy Institute has reviewed historical information regarding NRC Staff positions on this issue, and the results are included in this section. From a review of the testimony of Knight [C3], Maccary [C16], Atomic Safety and Licensing Board [C17] and Draft Regulatory Guide 1.121, it is clear that the primary requirements for margin of safety for steam generator tubes were:

1. tubes with detected acceptable defects should not be stressed during the full range of normal reactor operation beyond the elastic range of the tube material,
2. the factor of safety against failure by burst at normal operating conditions should not be less than three, and,
3. the margin against failure under postulated accident loads is consistent with the ASME Code (this includes both burst and plastic collapse).

Meeting Requirement 1 assures that for the full range of normal reactor operation (i.e., ASME Level A and B conditions), the maximum membrane stress in the degraded tube will never exceed the yield stress for the tube material. Normal operation, as used here, includes start-up and operational and upset transients that are included in the design specification for the steam generator. No safety factor is applied for evaluation of Requirement 1.

For Requirement 2, an explicit description of normal operating conditions that includes the full range of operational transients is not specified. However, Mr. Knight does point out that "... new steam generator tubes are typically manufactured with a wall thickness much greater than the wall thickness required by the design rules of ASME Code, Section III" and Mr. Knight further indicates that approximately twice the thickness (nominal/minimum) is typical. If in fact minimum wall calculations were based on the full range of operational transients or RCS design pressure safety limits, it would not be unusual for the margin above minimum wall to be on the order of 10 to 20% and not the margins suggested by Mr. Knight. In fact, the examples given in the above references use the normal operating differential pressure for the subject plant. Adopting a definition whereby a margin of 3.0 against gross failure or burst under normal operating conditions, including startup, operation in power range, hot standby, cooldown, and all anticipated transients that are included in the design specification, would require plugging limits to be lowered well below many licensees current NRC endorsed 40% plugging criteria.

With regard to Requirement 3, the intent of Draft Regulatory Guide 1.121 was to impose a margin against tube failure under postulated accident conditions consistent with the stress limits of Paragraph NB-3225 of ASME Section III, which covers faulted loads (i.e., Service Level D events). It was apparent that NRC staff guidance for tube integrity was focused on faulted conditions directly affecting the steam generators such as LOCA, steam line break, or feedwater line break. ASME Code Section III, Appendix F, which is for analysis of faulted load conditions, provides the basis for the 1.4 factor, derived from the allowable Level D limit of 0.7 S_u as previously discussed in Section 3.1.

Statements in Draft Regulatory Guide 1.121 provide additional insight for the industry's consistent application of this criterion. Section 3.a.1 of the Regulatory Guide states that loadings associated with normal plant conditions, including startup, operation in the power range, hot standby, cooldown and all anticipated transients, should not produce a primary membrane stress in excess of the yield stress. Once again, this statement is explicit, whereas the statements in Section 2.a.2 and 2.a.4 of the Regulatory Guide simply state a margin of safety under normal operating conditions should not be less than three. Further, in Paragraph C.2(6), Draft Regulatory Guide 1.121 imposes a margin against tube failure under postulated faulted conditions (Service Level D) consistent with the stress limits of Paragraph NB-3225 of the ASME Code, Section III, Subsection NB. This Code margin has been shown to be 1.4 based on load on primary membrane load, and 1.2 on combined primary membrane plus bending load.

C.5.2 Application of Industry Definition

NEI has reviewed multiple sources of information including sleeving topical reports, design and equipment specifications and licensee submittals in an effort to determine how loading conditions affecting steam generator tubing and pressure boundary components were assessed. This review considered application of the appropriate loading definitions, both during original design of components such as sleeves and plugs, and how the loading definitions have been applied during CM and OA of steam generator structural integrity.

C.5.2.1 Original Design

In establishing the safe limiting conditions for steam generator tubing, plugs and sleeves, the effects of loadings both at normal operation and postulated accident conditions are always evaluated. As such, all design requirements, specified in ASME Section III, Subsection NB, are applied to the initial design of RCS pressure boundary components. Consideration of primary membrane stress, bending loads, seismic, combined primary plus secondary stresses, and fatigue loads are addressed in the original design as required by the applicable Code paragraphs. Industry will continue to implement these requirements as appropriate for new steam generator designs and new repair methods proposed and submitted for NRC approval. However, as stated in Subsection NB-1110, "...(the ASME Section III rules) do not cover deterioration that may occur in service as a result of corrosion, radiation effects, or instability of the material." Instead, the performance of in-service steam generator components is assessed via CM and OA as required by NEI 97-06 and not the ASME Code.

C.5.2.2 Condition Monitoring

Regulatory Guide 1.160 [C18] and 10CFR50.65 [C1] state, in part, that licensees shall monitor the performance and condition of safety related components against established goals in a manner to provide reasonable assurance that such components are capable of fulfilling their intended functions. These goals are to be established commensurate with safety and, where practical, take into account industry-wide operating experience. In the case of steam generators, the performance criteria and definitions in NEI 97-06 provide the necessary protection against tube burst.

The information provided in this paper, has further demonstrated that the use of full power normal operating pressure differential is historically supported both by consistent industry application of the requirements and accepted regulatory positions. The use of this definition provides for uniform application, and is measurable and comparable within the CM process. While the factor of safety applied during the CM process is comparable with safety factors in the ASME Code, it should be noted that there is no Code requirement (either in Section III or XI) to conduct proof testing of components to be removed from service. Industry has, instead, adopted the accepted historical definitions described above as a basis for surveillance, testing and monitoring of steam generator performance.

For example, industry, through the development of the Steam Generator In Situ Pressure Test Guidelines [C8], has established the appropriate guidance for the verification of integrity performance through proof testing. In addition to a safety factor of three, the guidelines include adjustments to ensure representative testing. Typical adjustments include temperature, test equipment instrument error and bladder corrections. The guidelines further state: "That all appropriate loads should be considered for the damage form. For example, axial loads due to tube-to-shell temperature differences in OTSGs, or axial loading associated with locked tube supports in RSGs should be addressed to ensure test conditions are at least as severe as those expected during operating and accident events." These examples indicate that the use of industry's 3NOPD definition is conservative and appropriate to demonstrate structural integrity performance of degraded steam generator tubing.

C.5.2.3 Validation of Industry Definition

While the use of industry's 3NOPD definition facilitates the CM process, NEI has surveyed the NSSS designs to demonstrate that the application of the definition is bounding of Requirement 1 (or, Section 3.a.1 of Draft Regulatory Guide 1.121) for the full range of reactor operation.

As part of this survey, it was recognized that there are some design basis transients which exceed the normal steady state full power primary to secondary differential pressure for a limited time. A review of the design basis transients for CE, B&W, and Westinghouse designed plants is summarized in Table C-2. The table lists transients that may be experienced and exceed the normal full power differential pressure used in the 3NOPD definition.

Generally, analysis of these transients shows that if a degraded tube can withstand three times the normal full power differential pressure at operating temperature, then the stress due to these transients is less than yield at the corresponding temperature. For example, if the yield strength of tubes at the heatup/cooldown temperatures is greater than or equal to one-half of the ultimate tensile stress, and the differential pressure during heatup/cooldown is less than or equal to 1.5 times the normal operating pressure differential, then the performance criterion of 3NOPD is bounding as the no yielding criterion is satisfied. To comply with the requirements outlined in NEI 97-06, this conclusion should be demonstrated for each plant using plant specific conditions and material properties.

In summary, the industry 3NOPD definition has historical precedence, facilitates the CM process and is generally considered bounding. Application of the definition, coupled with the criteria and requirements of NEI 97-06 and supporting industry documents, provide reasonable assurance of steam generator tube integrity. More importantly, the structural and leakage criteria (including operational leakage) and assessment processes are measurable, achievable and provide appropriate defense-in-depth.

Table C-2
Typical Differential Pressures for NSSS Designs

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C.6 Assessment of Contributing Loads

In addition to the hoop and axial stresses induced by differential pressure in tubes of both RSGs and OTSGs, the tubes for both designs can be subjected to primary loads from dynamic events, and axial thermal loads resulting from tube-to-shell temperature differentials in OTSG designs, or similar loads associated with locked tube supports which could be postulated to develop in RSG designs. The basic design of the OTSGs involves straight tubes, which are welded to a tubesheet at each end. This gives rise to tube loads that do not typically exist in an RSG. For the RSGs, thermal axial loads can develop in the event that tubes become locked at tube support plate intersections. Consequently, the analytical procedures for assessment of degraded tubes and their repair hardware should account for this additional load, including consideration of appropriate safety factors.

C.6.1 Primary Loads

Contributing non-pressure primary loads during accident conditions would be those membrane and bending loads resulting from dynamic conditions. Dynamic conditions include cross flow, LOCA, and seismic events. In most situations, the inertial loads primary membrane loads are small. Dynamic tube loads, in of themselves, may not be major contributors to tube burst. In general, peak dynamic loads are of short duration and may not occur at the same time peak pressure and/or thermal differential loads are created. Therefore, tube burst will be controlled by the largest differential pressure conditions in most situations. If a transient analysis is used to calculate the magnitude of pressure and non-pressure contributing loads, then a time history would be evaluated to capture the time during the event that defines the limiting condition for tube integrity.

With regard to seismic loads and their contribution to net section plastic collapse, it has been shown in full size pipe tests that elastically derived loads well above material yield do not result in failure. Proposed revisions to the piping design rules have increased the stress limits from 3.0 S_m to 4.5 S_m or 6.0 S_m depending on the imposed loading combinations. Although not yet approved for new plant designs, these revised allowables would permit elastic stress limits to reach 1.5 to 2.0 times the ultimate tensile strength of the material. These limits were determined from pipe test studies.

Given the above discussion, the use of a safety factor of 1.2 for contributing primary loads from accident events is proposed as a way to keep the analysis for combined loads straightforward and conservative since only a single factor would be used. A single safety factor on combined primary loads is appropriate following the requirements of ASME Section III, Appendix F. From these requirements, a safety factor of 1.2 was derived for all sources of primary membrane plus bending loads.

C.6.2 Axial Membrane Loads in OTSG Tubing

The axial membrane load in OTSG tubes is made up of a combination of manufacturing-installed preload, load caused by difference in pressure between primary and secondary systems and temperature differences between the tubes and steam generator shell. The loads caused by differences in temperature have been shown to be the dominant load. Due to tubesheet flexure, the axial tube loads can vary significantly across the tube bundle, with the largest axial load typically occurring in the peripheral tubes.

Primary and secondary system pressures create axial load in the tubes in three ways. There is the load created by the differential pressure across the tubesheet, the loads resulting from pressure in the steam generator heads and on the steam generator shell (Poisson's effect), and Poisson's effect due to pressure differential across the tube wall. During normal steady state operation, the resultant pressure-induced load is typically less than a few hundred pounds tensile at peripheral tube locations.

The largest contributors of axial load in the OTSG tubes are the difference in temperature between the tubes and shell, and the difference in coefficient of thermal expansion between the tube and shell materials. Since the tubes have a larger coefficient of thermal expansion than the shell, even without a temperature difference between tubes and shell, the tube will experience a compressive thermal load under normal (steady-state) operating conditions when the tube and shell are at the same temperature. Coupled with a temperature difference that may develop between the tubes and shell created by more rapid heating or cooling of the tubes than the shell during a transient condition the axial load associated with thermal expansion can be significant.

In summary, the total axial tube load during normal steady-state operation will be a few hundred pounds compressive and vary slightly in magnitude across the bundle (from the center to the periphery). For normal operating design transients, the maximum elastically calculated tensile load occurs at the peripheral tube locations during the 100°F/hr cool down transient. The limiting elastically calculated compressive load during normal operating transients occurs during the 100°F/hr heatup transient.

The largest tube axial tensile loads occur during postulated accident conditions. Loads have been determined for Main Steam Line Breaks (MSLB), Small Break Loss of Coolant Accidents, and other plant specific conditions. The tube loads for the accident conditions are based on direct input from the power plant system thermal hydraulic analyses. These analyses are based on a combination of worst case scenarios that generally maximize the ΔP and ΔT between primary and secondary systems. Use of these values in determining tube loads results in what could be defined as conservative loads. Due to different transient response systems at each of the OTSG plants, the resulting accident tube loads are plant specific.

C.6.3 Axial Membrane Loads in RSG Tubing

The principal axial membrane load in RSG tubes is that created by differential tube pressure. However, in the event that a tube becomes locked at a tube support plate, it is possible that differential temperatures between the tube and the wrapper or other neighboring structures can create secondary axial loads in the locked tube. The magnitude of the axial loads is a function of the number of tubes that are locked and their location in the bundle. Locked tubes in the interior of the bundle, away from restraining structures such as the stay rods and wedge supports on the wrapper, will tend to displace the tube supports plates due to the flexibility of the plates resulting in little or no additional axial loads. This same type of behavior will occur during faulted conditions, resulting in little or no added stress for locked tubes away from restraining structures. A review of applicable loading conditions will indicate which tubes and under what conditions axial tube loads will become significant.

C.6.4 Treatment of Axial Thermal Loads

Axial thermal loads are classified as secondary for integrity evaluations of steam generator tube repair products and alternate repair criteria as discussed in Reference C18. The classification is based on the ASME definition in which secondary loads are self-limiting and by themselves will not cause failure under single load application. The axial thermal load is self-limiting and the repair products and alternative repair criteria are not subject to strain concentrations that could cause potential failure under a single load application.

For tube integrity evaluations, except for circumferential degradation, the axial thermal loads are classified as secondary. This methodology is also based on ASME Code definitions and test results. For example, burst testing of steam generator tubes containing axial degradation under applied axial loads from several hundred pounds compressive to nearly three thousand pounds tensile show the axial load has no effect on the burst pressure.

The classification of axial thermal loads for use in assessment of circumferential degradation will involve some level of engineering evaluation that will consider the magnitude of thermal load and the degradation mechanism under evaluation. As such, circumferential degradation will be evaluated on a case-by-case basis. The main condition is to verify that the axial thermal load falls within the definition of secondary loads for the given degraded condition.

The division between primary and secondary classification can be based on detailed analysis and/or testing. For example, tensile testing of steam generator tubes containing circumferential degradation show that for rather large crack like flaws (120 degree extent and 80% through-wall) the resulting elongation (2-inch gage length) of the tube prior to rupture is approximately 0.2 inches. This magnitude of elongation is sufficient to reduce the limiting thermal axial load in an OTSG tube by approximately 20%. In addition, the actual rupture load of the tube is greater than 3000 lbs. and approaching the limiting yield load of the unflawed OTSG tube cross section. For tube integrity evaluations, a comparison of the test results, elongations and rupture loads can be used to show adequate margin relative to the calculated design axial tube loads.

Review of the fatigue design rules for austenitic materials, the allowable alternating stress for ten loading cycles is approximately 70 ksi. This would be equivalent to full stress range of 140 ksi or about twice the flow stress for Alloy 600/690. For the case of non crack-like flaws, it appears from the fatigue design rules that failure in one application in load would not be expected. Therefore, one can generically classify axial thermal loads as secondary for non-planar flaws. Volumetric degradation represents a loss of material over a relatively large area compared to the tube cross sectional area of the tube. For such degradation, the local strain will occur over a finite region, which will be more able to accommodate axial deformations. Further, allowable stress range from fatigue design rules indicates failure of a tube in one load cycle is not likely. Therefore, for volumetric degradation, axial thermal loads can be considered as secondary loads.

When the axial thermal load is sufficiently high as to cause general tube yielding prior to burst at the degraded section, then the application of load will relax the applied load with increasing deformation. For this situation, failure from a single load cycle is not expected to occur. Therefore, the axial thermal load in the steam generator tubes is limited by the actual yield stress of the unflawed tube cross section (i.e., pure axial membrane stress). This is the absolute limit, with or without inclusion of a factor of safety.

C.7 Allowable Structural Limits

C.7.1 Tube Burst Condition

In applying the SIPC, structural limits for acceptable tube wall degradation are determined for two sets of conditions: one for steady-state normal operation and the other two for design basis accidents, as illustrated in Figure C-1.

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C.7.2 Plastic Collapse under Tension and Bending

Tube collapse by net section plastic failure under combined tension and bending loads is also to be evaluated to the SIPC. Plastic collapse is the condition where the tube fails by the formation of a plastic hinge under combined membrane plus bending loads and is not the situation where tube collapses due to external pressure. Tube collapse has been evaluated separately and found not to be a limiting condition compared with burst. Based on results from plastic collapse testing of flawed and unflawed tubing for both straight and U-bend geometries, it has been demonstrated that plastic collapse is not a relevant failure mode for steam generator tubing [C12]. For straight sections of tubing, substantial bending loads cause locking in tube support structures that restricts further tube end rotation and axial displacements. This effectively prevents plastic collapse in bending. Plastic collapse of U-bends under in plane bending leads to very large displacements. Such large displacements are restricted by interference with neighboring tubes. Therefore, tube burst is the most important failure mode for steam generator tube integrity for normal and accident loading conditions. No explicit assessment for collapse condition is required.

C.7.3 Circumferential Degradation

The combined condition of membrane and bending loads (Equation C-8c) may produce a limiting condition on acceptable indication size for circumferential degradation. In determining the structural limit for circumferential flaws, the maximum axial force from pressure load or total axial thermal load, for the limiting accident event, has been historically used to establish the acceptable degradation size for accident conditions. For OTSGs, the total axial force is dominated by thermal displacement-controlled conditions. Further, bending moments from cross flow and/or seismic conditions can be included if determined to be significant to tube burst. For this condition, the combined loads with appropriate safety factors are specified by Equation C-8c. It should be noted that typically, the maximum bending moments, largest differential pressure, and maximum differential temperature do not all occur at the same time in the transient. Such information can be considered in the determination of significant contributing loads.

When considering significant contributing loads, the structural performance criterion specifies that the condition for tube burst be established from combined primary loads plus axial stress from secondary loads for the appropriate condition. Contributing axial secondary loads during accident conditions are usually the result of differential thermal conditions. For OTSGs, the largest contributors of axial load in the OTSG tubes are the difference in temperature between the tubes and shell, and the difference in coefficient of thermal expansion between the tube and shell materials. Since the tubes have a larger coefficient of thermal expansion than the shell, even without a temperature difference between tubes and shell, the tube will experience a compressive thermal load under normal (steady-state) operating conditions when the tube and shell are at the same temperature. Coupled with a temperature difference that may develop between the tubes and shell created by more rapid heating or cooling of the tubes than the shell during a transient condition the axial load associated with thermal expansion can be significant. Due to different transient response systems at each of the OTSG plants, the resulting accident tube loads are plant specific.

The determination of structural limits for tubes in RSGs with circumferential degradation has been typically based on differential pressure. In the event that a tube becomes locked at a support plate, it is possible that temperature differences between the tube and the wrapper or other neighboring structures can create axial loads in the locked tube. The magnitude of the axial loads is a function of the number of tubes that are locked and their location in the bundle. During normal operation, the tubes that become locked will have a fixed axial stress, which is simply the stress due to the pressure load in an unlocked tube. Once locked, the axial stress may vary during transients that impose a differential displacement between the locked tubes and adjacent structures.

Locked tubes can occur in steam generators with carbon steel support plates due to corrosion of the plates. Since there is no available means to evaluate the extent of locking, one must assume all tubes are locked at support plates. Therefore, an analysis must be done to determine the effect of locked tubes on the loading in the U-bend. Available information indicates that tube locking does not occur in steam generators with broached hole, stainless steel tube support plates. Information regarding the application of a test pressure adjustment for locked tubing for CM proof testing is contained in the EPRI *In Situ Pressure Test Guidelines* [C8].

C.7.4 Axial Degradation

The presence of axial membrane or bending loads has been found to not have a significant effect on the burst pressure of tubes with axial degradation. The effect of axial membrane loads on burst was investigated by reviewing the basis for existing burst models. The current burst models have been fully validated to both pulled tube and laboratory test data. As a first approximation, the effect of varying biaxial stress on tube integrity was evaluated by modifying the burst relationship to account for the effect of axial stress to tube incipient failure. A von Mises' flow rule was assumed for yield behavior and it was assumed that axial load does not exceed the load to yield the tube. Under these assumptions, increasing the axial tensile load will not decrease the burst strength of the tube due to the increased constraint under biaxial tension. Therefore, for transients that cause tensile tube loads, the existing industry burst model for axial flaws are fully applicable.

For the situation when compressive loads are created, which could occur during some transients events (including some Level C events), a small reduction in burst strength is predicted from the simply theoretical model evaluated at the maximum differential temperature for a peripheral OTSG tube. Verification of these calculations has been conducted with experimental burst testing. These burst tests were performed on pressurized axially flawed tubes with varying axial loads from compressive to tensile. Test results showed that the burst pressure was not significantly different over the range of axial loads generated in OTSGs. Similar burst testing of axially flawed tubes with varying axial stress from bending loads also show no significant reduction in burst strength of RSG tubes. On this theoretical and experimental basis, the effect of axial load on burst strength of tubes with axial flaws will not be significant and therefore are noncontributing.

C.8 Summary and Conclusions

The use of normal steady state full power operating pressure differential has always been utilized as the basis for compliance with the 3NOPD criterion. The Technical Specification plugging limits for the plants surveyed were based on full power differential pressure. The surveyed design reports for minimum wall calculations for tubing and sleeves and Draft Regulatory Guide 1.121 compliance calculations were based on normal full power differential pressure. Additionally, use of the values listed in the design or equipment specification ensures consistency of application.

The recommended structural performance criterion is based on loading definitions and an evaluation framework consistent with the ASME Code and past regulatory guidelines. The use of a safety factor of 3.0 on normal steady state operation differential tube pressure, and 1.4 on the limiting design-basis accident differential pressure, follow from these historical practices.

Provisions are made for assessing additional loads when they are determined to be significant contributors to tube burst. This is expected to be the case in the assessment of circumferential degradation subjected to large primary bending loads and/or tensile axial secondary loads. This requirement will capture significant loading conditions such as those resulting from LOCA and seismic events, and thermal loads, such as those created in tube-to-shell differential expansions in OTSGs and RSGs with postulated locked tubes at a support plate. The application of axial membrane or bending loads does not have a significant affect on the burst pressure of tubes with axial degradation.

A safety factor of 1.2 on contributing combined primary loads is appropriate based on ASME Code Appendix F design rules. It has also been determined that this safety margin is consistent with ASME Section XI evaluation philosophy for piping. A safety factor of 1.0 for the contributing axial secondary loads is appropriate for use in the structural evaluation when they are also present. The technical basis for the use of a safety factor of 1.0 is that tube axial loads resulting from differential thermal expansion generally fit the classification of "secondary stress" with respect to ASME Code Section III design rules. The classification of thermal expansion stress as secondary stress has historically been made which is also consistent with ASME Code Section XI flaw evaluation acceptance criteria.

For tube integrity evaluations, except for circumferential corrosion degradation, the axial thermal loads are classified as secondary. This methodology is based on ASME Code definitions and test results. The treatment of axial thermal loads in assessing circumferential degradation should be evaluated on a case-by-case basis, which may be based on detailed analysis and/or testing.

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D

APPENDIX D: MODEL-ASSISTED POD DEVELOPMENT

D.1 Model-Assisted POD (MAPOD)

The motivation for MAPOD modeling is the economics associated with empirical POD approaches. For example, industry time and cost expenditures associated with acquiring experimental data for a recent EPRI facilitated performance demonstration in which multiple analysis teams analyzed bobbin and +Pt™ coil pulled tube data for axial ODSCC costs hundreds of thousands of dollars. Estimates for MAPOD predictions are on the order of an hour. These costs and time differences become even more pronounced when it is realized that the performance demonstration addressed only one steam generator tube wall degradation mechanism. With many other degradation mechanisms and noise sources yet to be considered, even larger industry costs would be imposed if this work was done experimentally.

Two MAPOD approaches are described in this appendix. The first is an analytical method referred to as Ahat modeling. It was developed during the early 1980s under Air Force funding as a part of their retirement-for-cause (RFC) program [D1-D3]. It has since become the paradigm for Department of Defense POD modeling with the issuance of MIL-HDBK-1823 [D4] and codified for industrial applications [D6-D7]. The second is a Monte Carlo approach developed as a part of the EPRI Tools for Tube Integrity Program [D7] in response to an NRC request that the nuclear industry develop a method for adjusting POD for noise.

Both MAPOD approaches described herein are predictive in their formulation. Ahat POD modeling is voltage based and as such does not treat noise explicitly. In addition, it does not address human factor effects providing an upper bound POD estimate analogous to a technique limit. The Monte Carlo approach is an extension of Ahat POD modeling. It treats both noise and human factors effects explicitly providing an estimate of system POD. Accordingly, it is a much more robust model. It has been implemented in an Excel™ workbook and is referred to as a Monte Carlo POD simulator [D7].

D.1.1 Ahat Modeling

Ahat modeling is a form of regression analysis in which a structural variable is used as the independent variable with signal amplitude used as the dependent variable. The dependent and independent variables are often transformed using $\log_e$-$\log_e$ or $\log_e$-linear transformations respectively to linearize the data. The dependent variable is always some measure of signal amplitude whereas the independent variable is a structural parameter such as length, depth, or whatever is of interest to the tube integrity engineer. With Ahat modeling, a continuous POD function is calculated directly avoiding the need for fitting a model to empirical binary hit_miss data. As stated previously, the resulting POD is analogous to a technique POD since it ignores whatever is of interest to the tube integrity engineer. With Ahat modeling, a continuous POD function is calculated directly avoiding the need for fitting a model to empirical binary hit_miss data. As stated previously, the resulting POD is analogous to a technique POD since it ignores data analyst human factor effects. The initial application of Ahat modeling was directed towards POD prediction for automated eddy-current analysis systems in which a threshold voltage is used to reduce false call rate. This threshold voltage is a parameter of the POD calculation so its impact on POD can readily be determined.

An example of Ahat analysis is shown in the upper graphic of Figure D-1 which shows a scatter plot for the natural logarithm of a structural variable with dimension "a" and the natural logarithm of the signal amplitude "$\hat{a}$". Paired data values are fit using a linear regression equation with intercept and slope values β_0 and β_1 respectively. The regression equation provides a model describing the paired data set (a, $\hat{a}$) predicting the mean value of $\log_e(\hat{a})$ for any value $\log_e(a)$ (solid line in the figure). The regression equation and its residuals (a measure of the scatter about the regression line) allow for the prediction of $\hat{a}$ for all values of a. Values of the structural variable a map into $\hat{a}$ in accordance with

The calculation of detection probability using regression analysis proceeds as follows. Imagine that a threshold voltage or decision threshold $\hat{a}_{dec}$ (horizontal solid line in the figure) has been established. To calculate POD(a_1), a normal distribution is centered at point a_1 along the regression line. POD(a_1) (shaded area in figure) is given by the integral of the residuals, modeled by a normal distribution, with lower limit of integration determined by $\hat{a}_{dec}$ Similarly, POD(a_2) is calculated by shifting the residual distribution to point a_2 along regression line. As before, POD(a_2) (shaded area in figure) is given by integral of a normal distribution with the lower limit of integration determined by $\hat{a}_{dec}$ Note that POD(a_2) > POD(a_1), and that the POD is a function of both structural and signal parameters. The lower graphic in the figure shows how POD(a_1) and POD(a_2) map into a POD function.

Reference D4 describes an equation which permits the direct calculation of POD as a function of the regression model equation parameters and the decision threshold. This equation is given by

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Figure D-1
Ahat POD Modelling

An example of how the ahat method can be used to estimate a technique POD using ETSS data is now considered. The Excel™ worksheet shown in Figure D-2 shows a listing for a set of degraded laboratory samples with corresponding metallographic depth and eddy-current amplitude measurements. For POD prediction, maximum depth is used as the structural variable while signal amplitude is measured using volts peak-to-peak using a P1 process channel (conventional support plate mix). A scatter plot of the raw data (columns B and C) is shown in the worksheet. These data are then transformed using the natural log function (columns D and E) and then regressed using the standard Excel™ regression function. The regression results are shown in the ahat worksheet shown in Figure D-3 with the information needed for ahat POD calculation highlighted. This information includes the regression equation slope, intercept and standard error. Figure D-4 shows the POD Calc worksheet in which Equations D-1 and D-2 have been programmed. The spreadsheet cell locations where the regression information is entered by the user are identified. The user then enters a threshold voltage in cell A2. Depressing the F9 key initiates an ahat POD calculation providing the user a plot as shown in the figure. For this example, a 0.5 volt threshold was used.

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Figure D-2
Excel™ Implementation of Ahat POD Modeling for Cold-Leg Thinning ETSS Data

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Figure D-3
Excel™ Implementation of Ahat POD Modeling for Cold-Leg Thinning ETSS Data

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Figure D-4
Excel™ Implementation of Ahat POD Modeling for Cold-Leg Thinning ETSS Data

D.1.2 Noise-Dependent Structural POD Modeling

As discussed previously, traditional POD models use a structural parameter such as degradation length or depth as the explanatory variable. Binary hit-miss data (NDE detections and non-detections) are plotted with the structural parameter as the independent variable. These data are then regressed with a non-linear function chosen as the detection probability model. See Figure 4-1. Since noise is not a model parameter, adjustments for noise cannot be readily made. Traditional POD models also ignore the fact that a data analyst observes *signals* and not the structural variable directly suggesting an incorrect choice or at best an incomplete set of explanatory variables. A more complete POD model would require coupled signal and structural variables with noise as a model parameter.

D.1.2.1 Ahat (S/N) Modeling

A key aspect of Ahat modeling is the use of what is referred to as "*â* versus *a*" or "Ahat analysis" for short. In this context, "*a*" represents a discontinuity dimension whereas "*â*" denotes an estimate based on measurement. For tube integrity applications, "*â*" is nothing more than a signal parameter such as voltage whereas "*a*" is a structural parameter such as depth or length. A scatter plot of transformed variables (*a*, *â*) is made, for which the two variables are coupled by regression analyses. This coupling is significant in that it provides a natural starting point for a predictive noise-dependent structural POD model.

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D.1.2 2 Monte Carlo Ahat (S/N) Simulation

Equation D-5 can be formulated as a Monte Carlo problem by replacing the variables ***a*** and N with distributions. Random sampling of the structural variable ***a*** gives a total signal amplitude *S* in accordance with Equation D-4. Sampling from a noise distribution *N* followed by division of the signal and noise components gives the (*S/N*). Thus, a Monte Carlo simulation generates many thousands of random combinations of $(S/N)_{MC}$ in which values of the structural variable ***a*** and (*S/N*) are coupled.

Figure D-5 shows input and output distributions for an Ahat Monte Carlo simulation. Degradation depth is normally used as the structural variable ***a*** for tubing. During Monte Carlo simulation, the structural variable ***a*** is sampled using a uniform distribution over all degradation depths ranging from 1-100% through-wall as shown in the figure. This sampling strategy provides a non-biased estimator of output (*S/N*). The lower graphic in the figure shows the Monte Carlo generated output (*S/N*) distribution fit using a log normal function. Since a conventional Ahat regression model in effect describes all possible signal amplitudes associated with a particular paired data set, its Monte Carlo formulation in terms of signal and noise amplitudes describes the population (*S/N*).

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Figure D-5
Monte Carlo Simulation of Ahat (S/N) Data

The Ahat (*S/N*) models for Equation D-5 are both tube degradation mechanism and eddy-current examination technique specific. The number of different Ahat models and noise distributions needed to address all the situations encountered in nuclear units with older steam generators that have numerous forms of tube wall degradation increases the complexity of POD model development for OA applications. The input noise distribution N generally is plant and tube location specific depending on the occurrence of secondary-side tube deposits, tube support structures, and tube geometry changes. Recently developed commercial eddy-current software is used to monitor tube noise at all locations throughout the length of a tube from which location specific noise distributions can be constructed. During one recent plant application, thirty separate POD models were developed and simulated using different combinations of Ahat data sets and noise distributions [D9].

D.1.2.3 Incorporating Human Factor or Personnel Effects

Another shortcoming of conventional Ahat modeling is that it does not address human factor analysis effects, i.e., missed indications, effectively only providing an estimate of technique POD. Since data analysts deal with signals embedded in noise, a basic signal detection theory first principles approach would describe the reliability of an analyst to report a signal using a reporting probability which is (*S/N*) dependent. Analyst reporting probability (ARP) has been modeled using both continuous and discrete functions as shown in Figure D-6.

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Figure D-6
Modeling Data Analyst Human Factor Effects using a (*S/N*) Dependent Reporting Probability

The continuous model uses a two-parameter log-logistic function with (S/N) as the explanatory variable. This function is described by the equation

From an overall NDE system POD modeling perspective, the link between tube wall degradation, eddy-current signals, and noise is now complete. In the context of appropriate explanatory variables, the model is defined by Equations D-5 and D-6. Table D-1 shows the output of a Monte Carlo simulation of these two equations. The table lists 1) trial value; 2) sampled value of the structural variable ***a*** (maximum depth in %TW); 3) associated Ahat signal amplitude; 4) sampled noise amplitude from a noise distribution *N*; 5) the *S/N* ratio obtained by dividing the signal and noise amplitudes; and 6) binary hit-miss data, i.e., 0 or 1, generated using an analyst reporting probability model.

Table 13-1
Example Monte Carlo POD Simulator Output Data

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The column two and column six entries from Table D-1 are treated as a conventional hit-miss data set. Although the hit-miss data were generated based on (*S/N*) considerations, the final detection probability model is generated using the structural variable ***a*** as the independent variable. Figure D-7 shows an example of Monte Carlo binary hit-miss data generated in accordance with Equations D-5 and D-6. Typically 10,000 simulations are run with the resulting hit-miss data regressed using a log logistic function. As shown in the following section, the resulting hit-miss data are a function of the input noise amplitude distribution *N* with the resulting POD model defined as the noise-dependent structural POD.

Monte Carlo POD modeling offers several advantages over empirical approaches which are often limited by available data both in terms of the number of data points and their distribution along the structural variable axis (See Figure 4-1 for an example of a sparse inadequately sampled data set). With Monte Carlo simulation, the structural variable can be sampled throughout its entire range with the sampling density controlled by the user.

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Figure D-7
Monte Carlo Generated Noise Dependent Structural POD Model

D.1.2.4 Illustrating the Dependency of POD on (S/N)

Loosely speaking, noise is defined as "any unwanted component of a signal." More specifically, those spectral components of a signal that do not carry useful information for the application at hand are referred to as noise. From a tube integrity perspective, noise might be considered as any signal not attributable to tube wall degradation. Noise signals (referred to as *N*) may perturb or mask flaw signals (referred to as *S*), precluding reliable detection by an analyst. Noise signals may also influence flaw sizing distorting the true flaw signal.

Figure D-8 shows four cumulative noise distributions used to illustrate the effects of noise on structural POD. The reference distribution represents a starting point with the other distributions scaled upwards and downwards by factors shown in the figure. Monte Carlo simulated noise-dependent structural detection probabilities corresponding to these distributions are shown on Figure D-6. Note that a structural POD is being shown i.e., the structural variable depth is used as the independent variable, with noise as a parameter.

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Figure D-8
Cumulative Noise Distributions Used for Noise-Dependent Monte Carlo POD Simulations

The curves in Figure D-9 illustrate the general sensitivity of structural POD to noise as its amplitude is scaled upwards or downwards. Intuitively, the curves shift relative to the reference noise POD in accordance with expectations. For cases in which the scaled noise amplitude distributions are greater than the reference distribution, the POD is shifted to the right of the reference POD indicating that POD is degraded. The opposite occurs for the two noise distributions less than the reference noise distribution, i.e., POD shifts to the left resulting in an improvement in detection.

D.1.2.5 POD Model Prediction and Validation

Noise-dependent structural POD model validation was accomplished using manual analysis results from five sets of eddy-current data for which metallographic ground truth was known. These data sets, consisting of approximately five-hundred grading units, were analyzed by ten teams of analysts (forty individuals) with a redundant analysis team structure duplicating current industry plant practices. All participating analysts were trained and provided data analysis guidelines in accordance with industry protocol.

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Figure D-9
Monte Carlo Simulated Noise-Dependent Structural Detection Probabilities Showing the Effects of Increasing and Decreasing Noise

Noise-dependent structural POD predictions were made for each of the five EPRI Performance Demonstration eddy-current data sets discussed in Section 4.2.2. Three basic inputs are required for prediction; 1) an Ahat data set as characterized by its regression equation, 2) a noise amplitude distribution, and 3) a model for data analyst reporting probability. Ahat models and noise distributions were constructed for each of the five data sets with structural variable sampling weighted based on the distribution of metallographic depths for each set. A step-function was used to model analyst reporting probability. Equations D-5 and D-6 were programmed into a Monte Carlo POD simulator used to make POD predictions prior to being given access to the manual analysis results.

Figure D-10 shows predicted POD; technique limit POD; and GLM weighted average POD for one of the validation test data sets. The Monte Carlo Simulator-predicted POD is shifted to the right of the technique POD as it should be reflecting human factor effects. It compares favorably with the multiple team GLM weighted average POD. Chi-square parameter analysis of the difference between predicted and experimental results shows the difference is statistically significant at a 5% level. However, most of the difference is for values of the structural variable < ~50% through-wall. Differences at these shallower depths are not expected to drastically impact OA predictions. Monte Carlo Simulator POD predictions have also been compared with other validation data set experimental results with comparable or better prediction accuracies.

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Figure D-10
Monte Carlo Predicted POD Compared with Technique Limit and Weighted Average POD for one the Performance Demonstration Datasets

D.1.2.6 Applications

Adjusting POD for Noise

An important objective of the EPRI Tools for Tube Integrity program was the development of a method to adjust POD for noise. Since the Monte Carlo POD simulator uses a noise distribution as one of its inputs for structural POD prediction, adjustments for noise are readily accomplished by replacing the initial noise distribution with a new noise distribution and simply recalculating POD.

In lieu of recalculating POD, Reference D6 recommends that the two cumulative noise amplitude distributions first be compared using a Kolmogorov-Smirnov (K-S) test. If the distributions are equivalent then there is no need to recalculate POD; otherwise, the POD should be recalculated. Figure D-11 shows two noise distributions identified as Baseline and Current Outage. The former is the result of an initial baseline noise examination whereas the latter represents the results from a subsequent outage. In accordance with [D6], distribution equivalence is accomplished using a two-sided (K-S) test. The result of this test shows that the two distributions are not equivalent which requires that the structural POD be recalculated.

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Figure D-11
Kolmogorov-Smirnov Comparison of Two Noise Distributions

POD recalculation is accomplished using the simulation logic shown in Figure D-5 with the baseline input noise amplitude replaced by the current outage noise amplitude distribution; all other inputs remain the same. Initial and recalculated PODs are shown in Figure D-12. For this example, the structural POD has improved slightly (shifted to the left of the initial POD) based on the current outage noise amplitude distribution.

A very common industry practice is to use a +Pt™ coil to confirm bobbin coil reported indications. Bobbin coil indications not confirmed by a +Pt™ coil are treated as NDD reducing the initial degraded tube population. Since two eddy-current examination techniques (bobbin and +Pt™) are being used sequentially, an effective or serial POD needs to be derived and used for an OA. At the outset, general statements can be made with regards to bounding values for this effective POD. For example, assuming that the +Pt™ POD performance is perfect, i.e., equal to 1 over the entire range of the structural variable, then the bobbin coil POD is the upper bound limit for the effective POD since all reported bobbin coil indications are confirmed by +Pt™. Since in reality, the +Pt™ coil POD $\lesssim$ 1 over some range of the structural variable, the effective POD will be less that the bobbin coil POD over this same range. That's basically all that can be stated.

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Figure D-12
Kolmogorov-Smirnov Comparison of Two Noise Distributions

+Pt™ Confirmation of Degradation (Deriving an Effective POD for an Operational Assessment)

The effective POD can be readily determined using the Monte Carlo simulation logic shown in Figure D-13. Two simulations are setup in parallel; one for the bobbin coil and another for the +Pt™ coil. The structural variable is assumed to be flaw depth in percent-throughwall. A random value is selected. For the same randomly selected depth, two signal amplitudes or voltages are then generated using Ahat models for bobbin and +Pt™ coils. Random samples are then drawn from noise distributions for both bobbin and +Pt™ coil. (S/N) ratios are then determined for both the bobbin and +Pt™ coil by dividing the signal and noise amplitudes. These *(S/N)* ratios are then processed using an analyst reporting threshold model generating a paired data set of bobbin coil and +Pt™ coil hit_miss values. The results are written to a file with the entire process shown in the figure repeated 10,000 times.

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Figure D-13
Simulation Logic for Deriving Effective POD

Three PODs can be calculated as shown in Figure D-13. Independent bobbin and +Pt™ PODs using all the hit_miss for each of the respective coils and an effective POD using +Pt™ hit_miss data from the bobbin coil detection subset. The various data sets are illustrated in Figure D-14 which shows Monte Carlo POD simulator outputs. Columns B and C and columns D and E are depth/hit_miss data for bobbin and +Pt™ coil respectively. Note that the depth values are the same for both coils. The upper graphic shows a portion of the full simulation output of 10,000 cases. The hit_miss data generated for each coil are independent (columns C and E) with a more frequent occurrence of hits for the bobbin coil than for the +Pt™ coil suggesting a better POD for the former rather than the latter. The lower graphic shows the output data sorted by bobbin coil hits simulating the case of bobbin coil detection to be followed by +Pt™ confirmation. The +Pt™ depth/hit_miss data in columns D and E are used to generate the effective POD. The three PODs are shown in Figure D-15; independent bobbin coil and +Pt™ coil PODs and the effective

POD. For the Ahat models and noise distributions considered in this simulation, the effective POD is less than the +Pt™ POD. So for degradation and OA applications, the +Pt™ POD would be used for tube integrity applications.

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Figure D-14
Simulation Outputs for +Pt™ Confirmation

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Figure D-15
Comparison of Effective POD with Bobbin and +Pt™ coil PODs (+Pt™ Confirmation)

D.2 References

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E

APPENDIX E: EXAMPLES OF CONDITION MONITORING AND OPERATIONAL ASSESSMENT LIMIT DETERMINATION

In this appendix example CM and OA calculations are presented. The types of degradation considered are:

- 100% Through-wall Axial Crack at a Freespan Location
- Circumferential Cracking with Restricted Lateral Tube Motion, 3ΔP Pressure and Accident Pressure Combined with Contributing Accident Loads
- Uniformly Deep 360° Thinning Over a Given Axial Length

Both deterministic and probabilistic approaches are used to compute CM and OA limits. Not every approach is illustrated. Other approaches should yield similar results.

E.1 Axial Cracking Examples

E.1.1 Example of Freespan, Through-wall Axial Crack

This example considers a through-wall axial flaw in a straight leg (assume Alloy 600 thermally treated).

Input Parameters:

Tube Diameter (D)	= 0.875 in.
Wall Thickness (t)	= 0.050 in.
Tube Inner Radius (R_i)	= 0.3875 in.
Tube Mean Radius (R_m)	= 0.4125 in.
Material Strength ($S_y + S_u$)	= 137.56 ksi
Sigma Value for ($S_y + S_u$), σ_M	= 6.3449 ksi
Pressure, 3 ΔPN_{Op}/1.4 P_{acc}	= 4.473 ksi (ΔPN_{Op} = 1491 psi)
NDE Technique Uncertainty in Length	
Systematic error (intercept), A_0	= 0.0
Systematic error (slope), A_1	= 1.0
Random error, $\sigma_{Technique}$	= 0.13 in.

NDE Analyst Uncertainty, $\sigma_{Analyst}$ = 0.06 in

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E.1.2 Structural Limit

E.1.3 Condition Monitoring Limit Using Arithmetic Method

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E.1.4 Condition Monitoring Limit Using Simplified Statistical Method

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E.1.5 Growth

E.1.6 Monte Carlo Analysis

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Figure E-1
Burst Pressure as a Function of Critical Crack Length for the Three Methods

E.2 Circumferential Cracking Examples

E.2.1 Circumferential Cracking with Restricted Lateral Tube Motion, Pressure and Bending Loads

E.2.2 Input Parameters

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E.2.3 Governing Equations

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Figure E-2
Burst pressure as a function of PDA for circumferentially cracked tubes [E1]

E.2.4 Limiting Structural Integrity Performance Criterion

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E.2.5 Pressure Only

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Figure E-3
Burst Pressure as a Function of Fractional PDA, NDE Reading, ID Cracking

Figure E-4
Burst Pressure as a Function of Fractional PDA (Physical), ID Cracking

E.2.6 Pressure Plus External Bending and Axial Loads

As discussed in Section *E.2.3* the effects of external bending and axial loads on the burst pressure of tubing with circumferential degradation has been quantified by an experimental program. The burst pressure for combined loading is obtained by subtraction of a burst pressure reduction term from the burst pressure equation for pressure only loading. For the U-Bend problem of interest, the ratio of R_B/R_m is greater than 52.5. Therefore the burst pressure reduction term is given by Equation E-14. It is subtracted from Equations E-10 and E-11 after they have been modified by Equation E-12 to account for ID cracking.

The calculation steps are the same as presented in Section E.2.5 except for the inclusion of one additional term for the burst pressure reduction due to combined external bending and axial loads. It should be noted that the bending and axial stresses are increased by the required safety factor of 1.2 in computing the burst pressure reduction term. The calculated burst pressure under combined loading must then meet the required safety factor of 1.2. Thus, after consideration of external loads, the limiting accident controlled SIPC burst pressure is 3180 psi (1.2*2650 psi accident pressure).

Following the format of Section *E.2.5* CM limits can be obtained from a plot of burst pressure versus PDA as an NDE reading. The NDE PDA values are first converted to a physical PDA values via the NDE regression sizing equation, Equation E-15, and then burst pressures are calculated following Arithmetic, Simplified Statistical and Monte Carlo approaches. The results are shown in Figure E-5.

The bounding Arithmetic CM limit is a PDA of 38.3 for a limiting SIPC pressure of 3180 psi. This accounts foe external loads and provides a factor of safety of 1.2 relative to the accident pressure. The Simplified Statistical CM limit at 95/50 is an NDE PDA value of 61.0. The Monte Carlo approach is somewhat more accurate for the large number of terms involved and leads to NDE PDA value of 61.5 for the CM limit. As is often noted the Arithmetic approach leads to very conservative CM limits.

Consideration of external loads makes accident conditions the limiting SIPC compared to pressure only loading. The CM limit for pressure only loading is an NDE PDA value of 65.6. When accident loads are considered this reduces to 61.5. A small difference is expected since the analysis of Section *E.2.4* shows that accident conditions are limiting by a small margin. There are specialized circumstances where accident conditions are significantly more severe than $3\Delta P_{Nop}$.

Figure E-6 shows plots of the best estimate (nominal or 50/50) burst pressure versus fractional PDA along with the Arithmetic bounding estimate of the 95/50 burst pressure and the 95/50 burst pressure calculated via both the Simplified Statistical and Monte Carlo approaches to combining uncertainties. The input is the actual physical PDA, not an NDE reading. Only material property and burst equation uncertainties are included. The resulting burst pressure for the Arithmetic, Simplified Statistical and Monte Carlo approaches is the actual physical PDA that results in the calculated burst pressure at 0.95 probability with 50% confidence. The PDA value when the calculated burst pressure is at the limiting SIPC is the end of cycle allowable PDA.

The EOC allowable limit for PDA to meet the required safety factor of 1.2 on accident loads is 53.2 for the Arithmetic approach, 59.6 for the Simplified Statistical approach and 59.7 for the Monte Carlo approach. As expected this is a modest amount smaller the pressure only $3\Delta P_{Nop}$ EOC allowable limit PDA value of 64.6. Applying an upper 95th percentile PDA growth allowance of 19.65, as in Section *E.2.5*, leads to the requirement that the worst case undetected PDA at the beginning of the cycle cannot be greater than 40.05 (59.7-19.65). Again this is not a problem in actual practice.

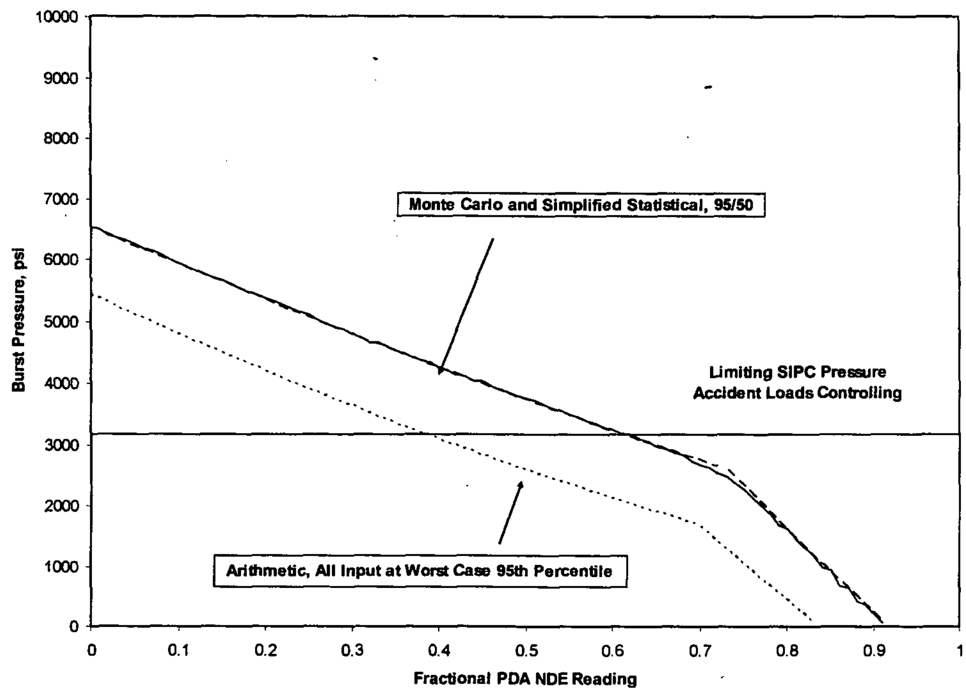

Figure E-5
Burst Pressure as a Function of Fractional PDA, NDE Reading, ID Cracking

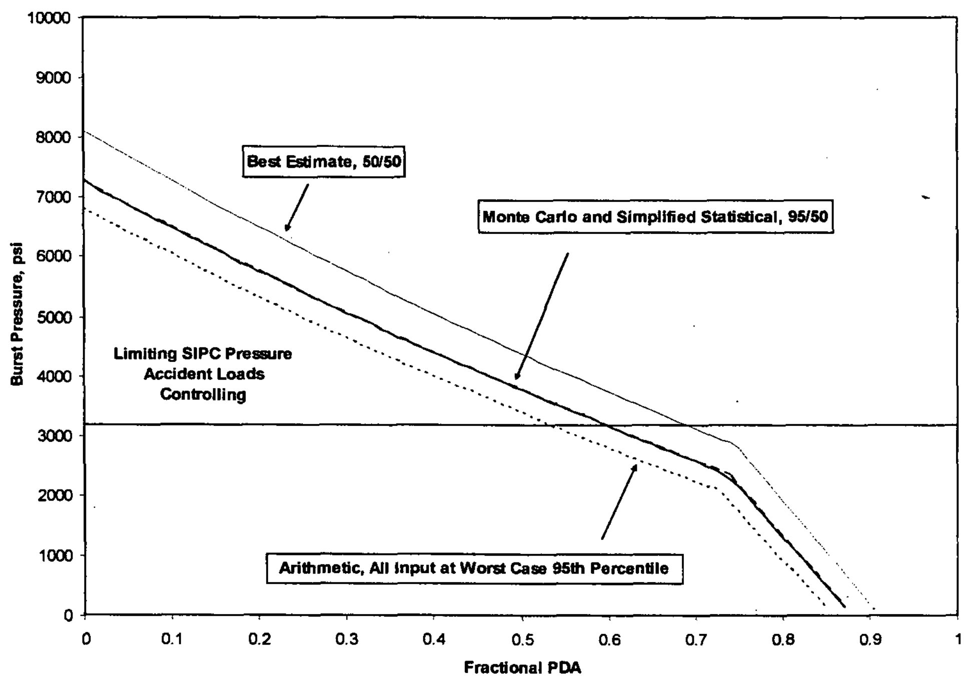

Figure E-6
Burst Pressure as a Function of Fractional PDA, (Physical), ID Cracking

E.3 Volumetric Degradation Examples

E.3.1 Example of Uniform 360° Thinning Over a Given Axial Length

Input Parameters

Tube Diameter (D)	= 0.875 in.
Wall Thickness (t)	= 0.05 in.
Tube Inner Radius (R_i)	= 0.3875 in.
Tube Mean Radius (R_m)	= 0.4125 in.
Material Strength ($S_y + S_u$)	= 137.56 ksi
Sigma Value for ($S_y + S_u$)	= 6.3449 ksi
Pressure, $3\Delta P_{oper}/1.4\ P_{acc}$	= 4.473 ksi (ΔPN_{op} = 1491 psi)
Length of Degradation	= 0.5 inch (Assumed equal to AVB width)
Error in Length (assumed)	
Systematic error (intercept), A_0	= 0.0
Systematic error (slope), A_1	= 1.0
Random technique error	= 0.07 in.
Analyst error	= 0.035 in.
Total random error, σ_{NDE}	= 0.078 in. (SRSS of technique and analyst errors)
	Error in Depth (ETSS 96004.3)
Systematic error (intercept), A_0	= 2.92 %TW
Systematic error (slope), A_1	= 0.96
Random technique error,	= 3.52 %TW
Analyst error	= 1.76 %TW
Total random error, σ_{NDE}	= 3.94 %TW (SRSS of technique and analyst errors)

E.3.2 Structural Limit

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Table E-1
Structural Limit Parameter h_{SL} Solutions for Several L Values

E.3.3 Condition Monitoring Limit

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Table E-2
Calculated CM Limits at the 95th Percentile

Table E-3
Calculated Burst Pressures at the 95th Percentile from Simplified Statistical Method

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E.3.5 Monte Carlo Analysis

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Figure E-7
Comparison of CM Solutions for a Burst Pressure of 4.473 ksi

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Figure E-8
Distribution of simulated burst pressures for a sample depth and length (1000 simulations)

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