EPRI

Impacts of the Structural Integrity Performance Criterion on Steam Generator Tube Integrity Evaluations

Technical Report

Impacts of the Structural Integrity Performance Criterion on Steam Generator Tube Integrity Evaluations

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REPORT SUMMARY

This report documents results of a study to determine the impacts of the revised structural integrity performance criterion (SIPC) on steam generator tube integrity evaluations. The report provides guidance for addressing loads other than pressure that must be considered in the revised steam generator tube SIPC of Nuclear Energy Institute (NEI) 97-06. Examples of load impacts are given.

Background

SIPC has been the subject of industry and Nuclear Regulatory Commission (NRC) discussions for several years, and several draft versions of SIPC have been proposed. The version addressed in this report was accepted by industry and NRC staff representatives on May 14, 2004. SIPC requires consideration of loading conditions associated with design basis accidents or combination of accidents in accordance with the design and licensing basis. These loads include axial and bending loads on steam generator tubes. Guidance for considering some of these loads has not previously been available. Current guidelines require consideration of loads other than pressure but lack detailed guidance or a recommended methodology for assessing other loads.

Objectives

- To provide a method for addressing all loading conditions as required by the revised SIPC.
- To assess the impact of the revised SIPC on the structural limits of tube degradation.
- To establish criteria for the magnitude of loads for which the traditional three-times-normal operation differential pressure would continue to govern.
- To discuss the impact of the revised SIPC on *in situ* testing criteria.

Approach

The project team identified types of potential tube degradation addressed by the revised SIPC. Tube loading conditions to be addressed also were identified. The team identified or proposed methods for considering each type of loading on each type of degradation. Several examples are given to illustrate the methods and to evaluate the impact of the revised SIPC on the structural limits developed.

Results

Current methods for assessing axial degradation are unchanged by the revised SIPC. A method for assessing circumferential degradation has been demonstrated. The impact of the revised SIPC is shown to be significant in only a small number of situations.

EPRI Perspective

NRC's expectation has been that loads other than pressure should be included in assessing tube rupture. In response, the NEI Steam Generator Task Force (SGTF) has defined a revised SIPC that considers non-pressure loads, and EPRI has carried out the impact study to determine if combined loadings due to pressure and transient conditions could reduce the burst pressure of tubes. The SGTF and EPRI efforts were done in parallel, and the NRC was kept informed of each effort's progress.

The revised SIPC provides a generic and universal criterion that applies to all steam generator designs and will be included in NEI 97-06 Revision 2 and in the-soon-to-be-published Revision 2 of the *Steam Generator Integrity Assessment Guidelines*. It also will appear in the soon-to-be-published Generic Licensing Change Package (GLCP) SGTF has produced a white paper that provides the rationale for the revised SIPC.

Keywords

Nuclear steam generators
Structural integrity performance criterion (SIPC)
NEI 97-06 implementation
Integrity assessment

ABSTRACT

An assessment of the impact of the proposed structural integrity performance criterion (SIPC) that was tentatively accepted by industry representatives and the NRC staff during a meeting held at the NRC offices on May 14, 2004, for the evaluation of degraded steam generator tubes in nuclear power plants was performed. The impact of the revised structural integrity performance criterion statement on condition monitoring and operational assessment limits with respect to currently accepted criteria was quantified. Uniform thinning evaluations are usually performed using the criteria of the ASME Code and explicitly consider the effects of additional loading conditions on the operating stresses. Comparisons are made against the design requirements of the ASME Code. The application of stress limits *vis-à-vis* load limits for the evaluation of degraded steam generator tubes is usually not directly meaningful and empirically verified models based on loads are used for the evaluation of degraded tube acceptability. Available data indicate that the burst pressure of tubes with axial cracks is not meaningfully affected by applied bending loads; however, the same is not necessarily true for tubes with circumferential cracks. Insight into the relationship between applied bending moments and the burst pressure of tubes with circumferential cracks was obtained from the results of a test program that was part of this project. Using this insight, the SIPC was applied to a number of example plants with significant seismic and postulated accident loads. The effect of tubes being locked into their supports, e.g., by corrosion products, was also addressed. The impact of the revised SIPC on *in situ* testing criteria was found to be significant in only a few situations. Although the application of a bending load will tend to reduce the structural limits, the reduction was found to be significant in only a few specific situations.

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1
INTRODUCTION

This report documents the results of a study to determine the impacts of the revised structural integrity performance criterion (SIPC) on steam generator tube integrity evaluations. The previous version of the SIPC appears in Revision 1 of NEI 97-06, "Steam Generator Program Guidelines," [1] which establishes the framework for managing the US Nuclear Industry's steam generator (SG) programs. It specifies performance criteria for addressing structural integrity, accident induced leakage and operational leakage of steam generator tubes. Supplementary information is also provided in the EPRI report on Steam Generator Integrity Assessment Guidelines [2]. The SIPC applies factors of safety both to the normal operating differential pressure and the pressure differential that would result in the event of a design basis accident that depressurizes a steam generator, and reads as follows in each document:

> *"Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation and a safety factor of 1.4 against burst under the limiting design basis accident. Any additional loading combinations shall be included as required by existing design and licensing basis."*

1.1 The Revised SIPC

Although this criterion is conservative, it has been the NRC's expectation, and is backed up by a requirement in the Steam Generator Integrity Assessment Guidelines [2] that loads other than pressure should be included in assessing tube rupture. In response, the NEI Steam Generator Task Force (SGTF) has defined a revised SIPC that considers the non-pressure loads and EPRI has carried out the impact study to determine if combined loadings due to pressure and transient conditions could result in a reduction in the burst pressure of the tubes. The SGTF and EPRI efforts were done in parallel, and the NRC was kept informed of the progress of each effort throughout.

The revised SIPC is as follows:

> *"All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination*

> *of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads."*

This definition, which provides a generic and universal criterion that applies to all steam generator designs, will be included in NEI 97-06 Revision 2 and in Revision 2 of the Steam Generator Integrity Assessment Guidelines. It will also appear in the Generic Licensing Change Package (GLCP). A White Paper, which has been produced by the SGTF to provide the rationale for the revised SIPC, is reproduced in Appendix A of this report.

The SGTF White Paper presents historical documentation and technical information to demonstrate that there has been a consistent industry and regulatory approach to defining structural integrity and associated margins against tube burst. It provides a link between the revised loading definition and ASME Code definitions and requirements, historical regulatory requirements, and industry protocols previously approved by the NRC Staff. It 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 ensures 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.

The following key changes have been made to the SIPC:

The limiting design basis accident conditions were expanded from just faulted conditions to include all design basis accidents.

Those loads that are found to significantly affect burst shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads that contribute significantly to burst.

Similarly, those loads that are found to significantly affect collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads. The definition of collapse as used in the SIPC, and agreed upon by the industry and the NRC staff representatives, is as follows:

"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."

For example, collapse occurs at the load corresponding to the ultimate tensile strength in a material tensile test.

1.2 Phase I Impact Study

The EPRI project was completed in two phases. Phase I was an impact study to determine how the revised SIPC would affect Condition Monitoring and Operational Assessment limits for various plants. Three vendor companies (B&W Canada, Framatome ANP, and Westinghouse) determined what the impacts of the non-pressure loads would be on their steam generators. In addition, major issues were identified that would have to be resolved before the revised SIPC could be implemented.

The Phase I impact study showed that there are no major problems for most plants. In general, structural limits remain controlled by differential pressure loading for most degradation modes and bundle locations.

1. For once-through steam generators (OTSG) and replacement once-through steam generator (ROTSG) designs there was no significant impact from the requirements of the revised SIPC, and only minor document updates may be required.
2. For Combustion Engineering recirculating steam generator (CE-RSG) designs there also was no significant impact (again minor document updates may be required).
3. For CE replacement recirculating steam generator (RRSG) designs, little or no impact was expected but a need to quantify conservatisms in design analysis was identified.
4. For Westinghouse recirculating steam generator (W-RSG) designs, there was no impact for most plants; however, for plants with high seismic loads a potential impact from bending loads was identified if there is circumferential degradation in U-bend region. In this case bending moments from seismic loads may reduce the structural limits. Locked tube scenarios may also result in lower structural limits.

It was further established that there is no industry model for computing burst condition when bending loads are present (i.e., combined axial and bending load case). It was clear from the study that the different approaches used by vendors produced significantly different burst results. A burst pressure model for combined membrane plus bending loads was proposed and applied in Phase I, but this model had not been validated. It was also determined that the design basis calculations may not be adequate (too conservative) for use with SIPC evaluations.

Two issues in particular were identified in Phase I that needed to be resolved in Phase II:

1. An Industry model was needed to account for the effects of combined membrane and bending loads on the burst pressure of damaged tubes; and
2. Existing plant design analyses may not contain sufficient detail to provide the load information needed to apply the revised SIPC.

1.3 Phase II: Revised SIPC Implementation

Phase II addressed the outstanding issues and provided a path for successful implementation of the revised SIPC. In order to develop the Industry model it was necessary to implement a testing program to measure the reduction in tube burst pressure as a function of the applied bending load. The resulting information led to the formulation of an empirical correlation that is simple to use, yet firmly supported by data.

The Phase II implementation procedure consisted of the following:

1. A test program that led to a validated burst pressure model for combined membrane and bending loads. This test program also addressed the collapse issue. Appendix B is a complete report on the test program, including the test configuration, the results obtained, and the validated burst pressure model.

2. A guide for obtaining and understanding the existing design basis information, (i.e., conservatisms in the design analysis calculations, availability of results required for burst calculations, etc.). Appendices C and D develop this guidance.
3. Methods for evaluating contributing loads and their significance to tube integrity. Appendices C and D describe these methoda and contain examples on how to evaluate these loads.
4. Screening limits for bending loads so plants not affected or with limited design information can apply criterion with minimal impact. These limits are discussed in Appendices B, C, and D.
5. Guidelines for in situ pressure testing when contributing loads are significant and lead to reduced structural limits. Guidance is provided in Appendices C and D.

Section 2 contains a summary of the burst and collapse tests. A detailed description is provided in Appendix B. An evaluation of the test results and the burst model that has been developed is summarized in Section 3. The details are provided in Section 7 of Appendix B. Results of the impact study are summarized in Section 4, for CE and Westinghouse U-tube steam generators, and for B&W Canada replacement steam generators. Details are provided in Appendix C for CE and Westinghouse steam generators, and Appendix D for B&W Canada replacement steam generators. An example of how the revised structural limit can be obtained is given in Section 4 of this report. The major conclusions from the test program and the impact study are summarized in Section 5.

2
BURST AND COLLAPSE TESTS

2.1 Burst and Collapse Test Results

2.1.1 Collapse Test Results

2.1.1.1 Straight Tube Collapse Test Results

2.1.1.2 U-Bend Collapse Test Results

2.1.2 Burst Test Results

2.1.2.1 Straight Tube Burst Test Results

Figure 2-1
Straight Tube Burst Pressure vs Bending Stress

2.1.2.2 Burst Test of U-Bend Tubes

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EVALUATIONS OF THE TEST RESULTS

3.1 Finite Element Analyses of Collapse and Burst Tests

3.1.1 Collapse Analysis for Tubes Loaded in Bending

3.1.2 Burst Test Simulations

3.2 Development of a Validated Burst Model

3.2.1 Development of the Empirical Model

3.3 Extending the Empirical Model to Other Tube Sizes

3.3.1 Finite Element Analysis of a Very Large Radius U-bend

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ASSESSING IMPACTS OF THE REVISED SIPC

This Section summarizes the work reported in Appendices C and D on assessing the impacts of the revised SIPC. Appendix C focuses on original steam generators designed by Westinghouse and CE, and Appendix D focuses on replacement steam generators designed by B&W Canada. These appendices provide information and methods to the utilities that:

increase understanding of design basis information,

can be used to evaluate contributing loads and assess their significance to tube integrity,

provide screening limits for bending loads such that plants with limited design information can apply the criterion with minimal impact, and

recommend criteria for in situ pressure testing when contributing loads are significant and lead to reduced structural limits.

In addition to summarizing the results, the Diablo Canyon cases for throughwall circumferential cracks are also described in Section 4.3 below, as examples illustrating how to calculate the revised structural limit using the contributing loads information and the test equations.

4.1 CE and Westinghouse Steam Generator Results

4.1.1 CE Steam Generator Results

4.1.2 Westinghouse Steam Generator Results

4.2 Replacement Steam Generator Results

4.2.1 CE Replacement Steam Generator Results

4.2.2 OTSG Replacement Steam Generator Results

4.2.3 Assessment of Impact on Catawba 1 Flaw Sizes

4.3 Diablo Canyon Example: Throughwall Circumferential Cracks

4.3.1 Structural Limit Considering Pressure Only

4.3.2 Effect of Moment and Axial Load: Unlocked Tubes

4.3.3 Effect of Moment and Axial Load: Locked Tubes

5
CONCLUSIONS

5.1 Plastic Collapse in Bending

5.2 Effect of Bending Loads on Burst Pressure

5.3 General Significance of Bending Loads on Burst Pressure

5.4 Impact of Revised SIPC on Specific Steam Generators

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GLOSSARY

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ABBREVIATIONS AND ACRONYMS

The following is a list of abbreviations and acronyms used in this report.

ASME	American Society of Mechanical Engineers
BWC	Babcock & Wilcox (B&W) Canada
CE	Combustion Engineering (now Westinghouse Electric Company)
CM	Condition Monitoring
CMTR	Certified Mill Test Report
DA	Degradation Assessment
FANP	Framatome Advanced Nuclear Power (now AREVA)
FLB	Feed Line Break
ID	Inside Diameter
kips	Kilo-pounds
LOCA	Loss of Coolant Accident
MSLB	Main Steam Line Break
NOP	Normal Operation or Normal Operating Pressure
NODP	Normal Operating Differential Pressure
NRC	United States Nuclear Regulatory Commission
OA	Operational Assessment
OD	Outside Diameter
ODSCC	Outside Diameter Stress Corrosion Cracking
OTSG	Once-Through Steam Generator
PDA	Percent Degraded Area
RSG	Recirculating Steam Generator
SIPC	Structural Integrity Performance Criterion
SLB	Steam Line Break
SSE	Safe Shutdown Earthquake
TSP	Tube Support Plate
TTS	Top of the tubesheet

TW	Throughwall
W	Westinghouse
WOG	Westinghouse Owners Group

8
NOMENCLATURE

English Letter Variables

F	Force
F_p	Axial force from the internal pressure,
F_z	Externally applied axial load, usually zero,
F_c	Axial force due to pressure on the crack flanks
F_r	Reaction load from the tube remaining area
k	Empirical factor for adjusting the
M	Moment
N	Normal distribution, e.g., N(0, •) is normal with mean 0 and standard deviation •
P	Internal pressure
P_b	Tube burst pressure
P_{bR}	Burst pressure reduction
P_{FLB}	Internal pressure associated with the FLB event (2650 psi)
P_{SLB}	Internal pressure associated with the SLB event (2400 to 2560 psi)
R_B	U-bend radius
R_i	Tube inside radius
R_m	Tube mean radius
R_o	Tube outside radius
t	Tube wall thickness
y	Axis normal to the centerline of a tube
z	Axis parallel to the centerline of a tube

Greek Letter Variables

α	Crack half-angle
$\Downarrow$	Angle to the neutral axis from the center of the crack
. P_{NOp}	Differential pressure during normal operation
ε	Statistical error
.	Factor relating the pressure loading on the crack flanks
σ	Standard deviation
σ_b	Axial stress due to bending of the tube
σ_p	Axial pressure stress in a tube
σ_f	Material flow stress
σ_p	Axial stress due to internal pressure
σ_y	Material yield stress
σ_u	Material ultimate tensile stress
.	Degraded area fraction (PDA/100)

9
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3. *Circumferential Cracks in Steam Generator Tubes: Structural Analysis Model and Integrated Burst Pressure Data Base,* EPRI, Palo Alto, CA: December 1997. TR-107618, Volume 1.
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A

APPENDIX A: WHITE PAPER ON DETERMINISTIC STRUCTURAL INTEGRITY PERFORMANCE CRITERION DEFINITION

(Prepared by the NEI Steam Generator Task Force (SGTF), November 2004)

1.0 INTRODUCTION

2.0 STRUCTURAL INTEGRITY PERFORMANCE CRITERION

2.1 Statement of Structural Performance

2.2 Definitions

2.3 Technical Discussion

2.4 Limits on Yield Strength

3.0 ASME CODE REVIEW

3.1 Minimum Wall Requirements

(1)

3.2 Primary Loads from Accidents Events

3.2.1 Appendix F Considerations

3.2.2 Section XI Pipe Flaw Assessments

3.3 Secondary Loads from Accident Events

3.3.1 Definition of Secondary Loads

3.3.2 Code Practice

3.4 Summary of Code Considerations

4.0 HISTORICAL REGULATORY PERSPECTIVE

5.0 ASSESSMENT OF CONTRIBUTING LOADS

5.1 Primary Loads

5.2 Axial Membrane Loads in OTSG Tubing

5.3 Axial Membrane Loads in RSG Tubing

5.4 Treatment of Axial Thermal Loads

6.0 ALLOWABLE STRUCTURAL LIMITS

6.1 Tube Burst Condition

6.2 Plastic Collapse Under Tension and Bending

6.3 Circumferential Degradation

6.4 Axial Degradation

7.0 SUMMARY AND CONCLUSIONS

REFERENCES

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B

APPENDIX B: EFFECT OF BENDING STRESS ON BURST PRESSURE OF STEAM GENERATOR TUBES CONTAINING CIRCUMFERENTIAL DEGRADATION

Final Report, September 2004

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ABSTRACT

The nuclear industry has been working for several years on establishing a program to address the complete care of the steam generators. The program includes everything from inspection techniques and intervals to repair criteria. One of the open items to be resolved is the completion of specific requirements to be considered in the evaluation of structural integrity of steam generator tubing. The general requirements for structural evaluations have been outlined in a statement referred to as the "Structural Integrity Performance Criteria" (SIPC). The SIPC defines the various types of loading conditions that must be considered and the appropriate factors of safety that must be applied.

One of the last unresolved issues prior to getting the SIPC approved by the Nuclear Regulatory Commission (NRC) is the treatment of bending loads (contributing loads) and their appropriate safety factor. Although bending loads have been considered in past tube integrity evaluations, there is no specific test data to benchmark (validate) the existing bending failure models. In order to benchmark the models, SIPC program representatives have contracted AREVA to perform the necessary benchmark testing. This document provides the results of that testing program.

ABBREVIATIONS & ACRONYMS

BWC	B&W Canada
BWOG	B&W Owner's Group
CMTR	Certified Material Test Report
EDM	Electrical Discharge Machining
FANP	Framatome Advanced Nuclear Power, an AREVA company
FE	Finite Element
ID	Inside Diameter
MSLB	Main Steam Line Break
NRC	Nuclear Regulatory Commission
OD	Outer Diameter
OTSG	Once-Through Steam Generator
PDA	Percent Degraded Area
RSG	Re-circulating Steam Generator
SG	Steam Generator
SIPC	Structural Integrity Performance Criteria
SSE	Safe Shutdown Earthquake
Su	Ultimate Strength
Sy	Yield Strength
TSP	Tube Support Plate
TW	Through-Wall

DEFINITIONS

BURST 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. The definition is taken from Appendix M of Rev 1 of Steam Generator Integrity Assessment Guidelines, EPRI Report TR-107621-R1, March 2000

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." The definition is taken from an industry presentation to the NRC on May 14, 2004 on the status of the SIPC and will be incorporated into Tech Spec Bases.

1
INTRODUCTION AND BACKGROUND

1.1 Purpose

The purpose of this document is to provide the results of an EPRI sponsored program to determine the effect bending loads have on the burst pressure of steam generator (SG) tubes containing circumferential degradation. The program involves burst testing of tubes containing circumferential degradation with applied bending loads for both Once-Through Steam Generator (OTSG) and Re-circulating Steam Generator (RSG) tube configurations. In addition, a secondary task was executed to address the potential for bending collapse of both OTSG and RSG tubes.

1.2 Background

The nuclear industry has been working for several months on defining specific requirements that must be considered in the evaluation of structural integrity of steam generator tubing. The general requirements for the evaluation are being outlined in a statement referred to as the "Structural Integrity Performance Criteria". The SIPC defines the various types of loading conditions that must be considered and the appropriate factors of safety that must be applied.

One of the last items to be resolved is the treatment of bending loads (contributing loads) and their appropriate safety factor. Although bending loads have been considered in past tube integrity evaluations, there is no specific test data to benchmark (validate) the existing bending failure models. AREVA has been contracted to perform a testing program to benchmark the existing bending failure models and if necessary define a new bending failure model. This document provides the results of that testing program.

2
PROGRAM DESCRIPTION

3
TEST PARAMETERS

3.1 Tube Material

3.2 Design Bending Loads

3.3 Test Configurations

45.5” Straight Tube

Large U-Bend

Small U-Bend

3.4 Flaw Geometry

3.5 Test Loads

Figure 3-1
FE Model of 36" U-Bend

Figure 3-2
FE Model of 45.5" Straight Tube

4
SAMPLE PREPARATION

4.1 Tube Material

4.2 Forming U-bends

5
TEST EQUIPMENT

6
TEST RESULTS

7
EVALUATION OF TEST RESULTS

8
CONCLUSIONS

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C

APPENDIX C: SCOPING STUDY ON THE IMPACT OF NEI 97-06 STRUCTURAL INTEGRITY PERFORMANCE CRITERION

Consideration of Loads in Addition to Pressure for Recirculating Steam Generator Tubes

Final Report, September 2004

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REPORT SUMMARY

This report provides guidance for addressing loads in addition to pressure that are required to be considered in the revised steam generator tube Structural Integrity Performance Criterion (SIPC) of NEI 97-06. Examples of the impact of these loads are given.

Background

The Structural Integrity Performance Criterion has been the subject of Industry and NRC discussions for several years, and several draft versions of the SIPC have been proposed. The version addressed in this report was accepted by industry and NRC staff representatives on May 14, 2004. The SIPC requires consideration of loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis. These loads include axial and bending loads on the steam generator tubes. Guidance for consideration of some of these loads has not previously been available.

Objectives

To provide a method for addressing all loading conditions as required by the revised SIPC.

To assess the impact of the revised SIPC on the structural limits of tube degradation.

To establish criteria for the magnitude of loads for which the traditional three times normal operation differential pressure would continue to govern.

To discuss the impact of the revised SIPC on *in situ* testing criteria.

Approach

The types of potential tube degradation addressed by the revised SIPC are identified. Tube loading conditions to be addressed are also identified. Methods for consideration of each type of loading on each type of degradation are identified or proposed. Several examples are given to illustrate the methods and to evaluate the impact of the revised SIPC on the structural limits developed.

Results

Methods for the assessment of axial degradation are readily available. A method for the assessment of circumferential degradation based on previous EPRI sponsored research has been proposed and demonstrated. The impact of the revised SIPC is shown to be significant in only a small number of situations.

ABSTRACT

An assessment of the impact of the proposed structural integrity performance criterion (SIPC) that was tentatively accepted by industry representatives and the NRC staff during a meeting held at the NRC offices on May 14, 2004, for the evaluation of degraded steam generator tubes in nuclear power plants was performed. The impact of the revised structural integrity performance criterion statement on condition monitoring and operational assessment limits with respect to currently accepted criteria was quantified. Uniform thinning evaluations are usually performed using the criteria of the ASME Code and explicitly consider the effects of additional loading conditions on the operating stresses. Comparisons are made against the design requirements of the ASME Code. The application of stress limits *vis-à-vis* load limits for the evaluation of degraded steam generator tubes is usually not directly meaningful and empirically verified models based on loads are used for the evaluation of degraded tube acceptability. Available data indicate that the burst pressure of tubes with axial cracks is not meaningfully affected by applied bending loads; however, the same is not necessarily true for tubes with circumferential cracks. Insight into the relationship between applied bending moments and the burst pressure of tubes with circumferential cracks was obtained from the results of a recent test program. Using this insight, the SIPC was applied to two example plants with significant seismic and postulated accident loads. The effect of tubes being locked into their supports, e.g., by corrosion products, was also addressed. The impact of the revised SIPC on *in situ* testing criteria was found to be significant in only a few situations. Although the application of a bending load will tend to reduce the structural limits, the reduction was found to be significant in only a few specific situations.

1
INTRODUCTION

Revision 1 of NEI 97-06, "Steam Generator Program Guidelines," Reference 1, establishes the framework for managing steam generator (SG) programs, and specifies the performance criteria for addressing the structural integrity, accident induced leakage and operational leakage of steam generator tubes. In response to regulatory questions regarding whether all loads that could potentially affect the burst capability of tubes were considered, an industry study was sponsored by EPRI to review the normal operating and accident loading conditions and to determine if the combined loading due to pressure and transient conditions could result in a reduction in the burst capability of the tubes. This report summarizes the results of the program, which was performed in two phases: 1) Evaluation of membrane and bending stresses and secondary stress effect on tube burst capability, and 2) update of the initial analysis based on specific test data provided by others to determine the magnitude of the accident loads on the burst capability, and development of specific instructions for recovering the necessary loads from plant-specific design documents. The report integrates the

1.1 Background

The objective of this study was to address a revision to the structural integrity performance criteria (SIPC) statement. The SIPC as delineated in NEI 97-06 is:

> *"Steam generator tubing shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation and a safety factor of 1.4 against burst under the limiting design basis accident. Any additional loading combinations shall be included as required by existing design and licensing basis."*

Various revisions of the SIPC have been proposed for clarification and to ensure consistent application of the criteria. A preliminary evaluation, designated as Phase 1, of the impact of the changes to the SIPC was documented in Reference 2. The Reference 2 report concluded that the most severe loading conditions to be addressed for the bounding plant analyzed, Diablo Canyon, were feedwater line break in conjunction with the specified safe-shutdown earthquake (SSE). The revision of the criterion statement that is addressed in this report, designated as Phase 2, was tentatively accepted by industry and NRC staff representatives during a meeting held on May 14, 2004, Reference 3, and is nearly the same as that considered in Reference 2[1]. The final draft of

[1] The version of Reference 2 was discussed with the NRC staff. Changes were made during the Phase 2 program, leading to the final version as quoted in this report.

the criterion statement was subsequently transmitted to industry representatives for review and comment via Reference 4 as follows:

> *"All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads."*

The definition of collapse as used in the SIPC, and agreed upon by the industry and the NRC staff representatives, also per Reference 4, is as follows:

> *"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."*

For example, collapse occurs at the load corresponding to the ultimate tensile strength in a material tensile test.

The following key changes are noted from the rewording of the SIPC statement from that of the Reference 1:

The limiting design basis accident conditions were expanded from just faulted (Level D) conditions to include all design basis accidents (Levels C and D). It is noted that the original documented regulatory basis for the evaluation of the integrity of degraded SG tubes, that is, draft Regulatory Guide (RG) 1.121, does not include specific provisions for the treatment of Emergency (Level C) loading conditions as their severity with regard to the potential for bursting the SG tubes was assumed to be bounded by the Faulted (Level D) loading conditions.

Those loads that are found to significantly affect burst shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

Similarly, those loads that are found to significantly affect collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

1.3 Report Organization

In order to carry out the required evaluations the report was organized to address the structural limits and the Phase 1 model. Sections 2 discusses the definition of the structural limit and discusses the impact of this study on tube integrity assessments (Degradation Assessments and Condition Monitoring and Operational Assessments). Section 3 analyzes the relationship of tube degradation to burst. The identification of loads affecting SIPC compliance is provided Section 4, and the identification of load information from SG reports is considered in Section 5. Section 6 covers the calculation of SG tube structural limits using Diablo Canyon and Catawba 2 as specific examples, using a updated model developed from test data provided by another project participant. Section 7 addresses consideration of in situ testing to account for the additional loads, and Section 8 develops the threshold bending moment that will determine if non-pressure loads need to be considered in the calculation of the structural limit.

1.4 Project Overview

The objective of the first task was to specify how a plant can extract loads data from existing stress reports and document the basis for understanding of the adequacy of existing design analyses to provide the required accident loads affecting tube integrity is needed. A review of the stress reports for key plant SG designs was performed, to include Diablo Canyon and Catawba 2 as a minimum. The following aspects of the review process were performed to address the following subtasks:

1. Identification of Loads that Significantly Influence Burst—Identify the loads other than pressure loads that can influence the allowable level of tube degradation with regard to meeting the SIPC. The intent of this task was to utilize experience from the SIPC Phase 1 program of Reference 2 and to include additional evaluations beyond that available from Phase 1 activities, namely the data from the burst tests of circumferentially cracked tubes of Reference 6. The results of this evaluation are provided in Section 4 of this report.
2. Review of Existing Documents for Necessary Loads—Review existing design documentation, such as stress reports, provided with the SGs to determine if the necessary loads are available in the design documentation. Since no generic analysis method was utilized during the design of the SGs, there is no assurance that examination of only a sample (i.e., Diablo Canyon and Catawba) of the reports will provide a definitive answer to this question. Therefore, a larger sample (15) of the available reports was examined for both Westinghouse and CE design SGs for all models and vintages of the designs, with greater emphasis on those plants without a RG 1.121 analysis, to provide reasonable input as to the availability of the necessary loads in the existing reports.
3. Process for Identifying the Necessary Loads—Develop a process to obtain the necessary loads, whether, or not, they are available in existing design documentation. For necessary loads that are NOT available in existing design documentation, the analysis requirements to provide the loads are specified. A possible recommendation resulting from this effort may be to perform, or repeat, the seismic and LOCA analyses for specific plants. The plants requiring this from among the 15 plants reviewed are identified.

It is not possible to determine the level of conservatism in existing loads without significant effort for several reasons. Some plants were analyzed based on generic seismic spectra. These are construed to be "bounding" inputs to the analysis. Other plants were designed using plant-specific seismic spectra. While the analytical models are believed to be inherently conservative, consistent with good design practice, the actual factors of safety cannot be identified. For the plants designed using generic input loading, the only method available to identify the degree of conservatism is to repeat the analyses using plant-specific input loads.

4. Demonstration Example—Using Catawba Unit 2 and Diablo Canyon data as examples, demonstrate the process of subtasks 1-3 above. It is recognized that a potentially significant limitation of this approach is that these two plants may not be representative of the larger population of plants. For example, a significant level of analysis work has been performed since the original design analyses for both plants so that the design information required for SIPC is generally available.

2
STEAM GENERATOR TUBE STRUCTURAL LIMITS

3
RELATIONSHIP OF TUBE DEGRADATION TO BURST

support. Relative to the effects on axial cracks, it was noted in Reference 9 that bending stresses at TSP will not affect the tube burst strength so long as the bending stress is less than 34,000 psi.

4
IDENTIFICATION OF LOADS AFFECTING SIPC COMPLIANCE

This section deals with the identification and consideration of non-pressure loads that may affect the level of degradation that is in compliance with the structural integrity performance criteria. The intent is to assess the conditions under which the limiting level of circumferential degradation will be less for 1.2 times the combined postulated accident differential pressure plus imposed bending moment than for three times the normal operation differential pressure. The section treats Westinghouse and Combustion Engineering designed SGs separately.

5
LOAD INFORMATION FROM SG REPORTS

6
CALCULATION OF SG TUBE STRUCTURAL LIMITS

7
IN SITU TEST CRITERIA

8
BENDING MOMENT LIMITS

9
CONCLUSIONS

10
GLOSSARY

11
ABBREVIATIONS AND ACRONYMS

12
NOMENCLATURE

13
REFERENCES

14
ATTACHMENTS

D

APPENDIX D: IMPACT OF NEW STRUCTURAL INTEGRITY PERFORMANCE CRITERION ON REPLACEMENT STEAM GENERATOR TUBE INTEGRITY LIMITS

Phase 1 and 2 Summary Report

Final Report, October 2004

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ABSTRACT

This report summarizes a scoping assessment done to evaluate the impact of revised structural integrity performance criterion on the tube flaw size limits in nuclear steam generators. The assessment is performed for a bounding U-tube replacement steam generator design (RSG) and a bounding replacement once-through steam generator design (ROTSG). The assessment of a Westinghouse plant RSG is performed as well. Tube loads are extracted from the Design reports and, in some cases, recalculated to reduce the conservatism. Three bounding generic flaw types are considered, namely uniform wall thinning and two types of circumferential flaws. The basis for the structural integrity performance criterion is discussed, both with respect to the existing flawed tube analysis following NRC Regulatory Guide 1.121 (RG 1.121) and to the new criterion. The result of tube assessments based on the revised structural integrity performance criterion is compared to the tube performance limits. Since the most severe defect is shown to be the throughwall circumferential crack, the effect of combined bending moment and pressure loads on this flaw type is investigated with finite element and classical methods. The impact of the new criterion is examined in detail. To facilitate the extension of the results to other steam generators of similar design, the report includes a discussion of the factors contributing to the trends seen in the results. The potential impact on the operation of the stations is assessed. The report also includes discussions on plant-specific screening for the impact of the revised SIPC and on changes to in situ pressure testing criteria.

1
INTRODUCTION

The current industry-accepted structural integrity performance criterion (SIPC) for steam generator (SG) tubing is contained in Revision 1 of NEI-97-06 [1], with supplementary information provided in the EPRI report on Steam Generator Assessment Guidelines [2]. The regulatory position in the US has progressed from NRC Regulatory Guide 1.121 [3] to Draft Regulatory Guide 1074 [4], which proposes that loads other than pressure be included in assessing tube rupture. In the light of these developments, the SIPC has undergone some revisions to address regulatory and industry questions, and to clarify the application of the rules. The newly proposed criterion is given below:

> *"All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads."*

The rewording includes the following key changes:

a) Limiting accident conditions were expanded from just faulted conditions to all design basis accidents,

b) For treatment of non-pressure accident loads, a third requirement has been explicitly added whereby tube burst shall be assessed under a combined load basis with a factor of 1.2 on primary loads (PL) and 1.0 on axial secondary loads (ASL) that contribute significantly to burst,

c) An additional stipulation has been discussed in the definition of contributing loads where the axial secondary system loads are to be determined under elastic material behavior. This requirement has been dropped from the criterion.

The present report investigates the impact of the changes in the SIPC on the existing tube integrity analysis for Babcock & Wilcox Canada (BWC) Replacement Steam Generators. One bounding U-tube (CE type plant) and one once-through steam generator design are investigated fully. A partial analysis is performed for another (Westinghouse plant) U-tube design. The review includes the tasks listed below:

A general review of plants was performed to identify a specific plant or SG design that would be most adversely affected by the new SIPC.

The plant designs are selected and generic bounding flaw geometries are used for assessing the impact of new requirements.

For the most bounding circumferential through-wall cracks under mixed pressure and bending loads, the structural integrity of typical tube arrangements is analyzed and a comparison to experimental data and proposed models is made.

The loading conditions are identified and an assessment is performed to identify contributing non-pressure loads (e.g. seismic, cross-flow, etc.). The extraction of these loads from the design reports is discussed for two (U-tube) examples, and some loads are recalculated to avoid some of the conservatism in the Design calculations.

For the two bounding SG designs, scoping calculations determine whether changing the criterion to the new wording, specifically 1.2PL+1.0ASL, will impact existing past analyses or future methods of analysis. If there is any impact to structural calculations, the aspect of the new criterion that caused the impact is explained.

The impact of including non-pressure loads on in-situ pressure testing is discussed.

A list of factors that would make a plant more susceptible to reanalysis and/or backfit is developed (e.g., high seismic loads, etc.).

An approximate cost impact (cost of new/added analysis) is provided for the worst-case scenario, as well as for the typical case and its approximate cost for review/analysis.

The assessment is discussed, and calculations, significant findings, and conclusions regarding the impact of the new SIPC are summarized.

2
BASIS OF THE FAILURE ANALYSIS OF DEGRADED STEAM GENERATOR TUBES

Figure 2-2
Stepped Bar Illustrating Strain Concentration at a Short Flaw in a Long Tube

Figure 2-3
Behavior of Stepped Bar Under Axial Load, a) Moderately Weakened Section, b) Severely Weakened Section

3
EVALUATION OF TUBES WITH CIRCUMFERENTIAL THROUGH-WALL CRACKS UNDER COMBINED PRIMARY AXIAL AND BENDING LOADS

4
TUBE DEGRADATION ANALYSIS FOR REPLACEMENT U-TUBE STEAM GENERATORS

5
ASSESSMENT OF IMPACT OF REVISED STRUCTURAL INTEGRITY CRITERION ON U-TUBE STEAM GENERATORS

6
TUBE DEGRADATION ANALYSIS FOR REPLACEMENT ONCE-THROUGH STEAM GENERATORS

7
ASSESSMENT OF IMPACT OF REVISED STRUCTURAL INTEGRITY CRITERION ON ONCE-THROUGH STEAM GENERATOR TUBE

8
IMPACT ON IN-SITU PRESSURE TESTING

9
CONCLUSIONS

10
REFERENCES

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