NRC INSPECTION MANUAL

INSPECTION MANUAL CHAPTER 0609 APPENDIX J

STEAM GENERATOR TUBE INTEGRITY FINDINGS
SIGNIFICANCE DETERMINATION PROCESS

Effective Date: 05/18/2026

0609J-01 INTRODUCTION

This significance determination process (SDP) is most often used in conjunction with Inspection Procedure (IP) 71111.08, “Inservice Inspection Activities,” to estimate the risk significance of steam generator (SG) tube integrity issues that may result in failures to meet licensing bases and regulatory commitments as identified through the in-service inspection program.

This SDP can also be entered from the SDP screening questions in Exhibit 1 of IMC 0609 Appendix A, “The At-Power Significance Determination Process,” as a result of a degraded SG tube condition involving at least one tube that does not meet the structural integrity performance criteria (SIPC) (i.e., it cannot sustain three times the differential pressure across a tube during normal full power, steady state operation ($3\Delta P_{NO}$)) or one or more SG tubes that violate “accident leakage” performance criterion (i.e., involve degradation that would exceed the accident-induced leakage performance criterion (AILPC) under design-basis accident conditions). This chapter provides generic guidance for assigning the preliminary significance to inspection findings when SG tube degradation has exceeded tube integrity performance criteria. Table 1, Steam Generator Tube Integrity SDP Matrix (in Section 0609J-03 of this IMC), presents the guidance for determining the preliminary significance of SG tube integrity findings.

This SDP is not suitable for assigning significance to findings that involve only programmatic deficiencies in the licensee’s SG tube integrity inspection program and have no resulting consequence on actual physical tube integrity. If a programmatic deficiency causes an inspection inadequacy, current guidance considers this as a minor performance deficiency unless degradation of SG tube integrity beyond the structural integrity performance criterion (SIPC) or accident induced leakage performance criterion (AILPC) is demonstrated. If a programmatic deficiency is identified during the inspection, the inspector shall notify the NRC headquarters technical staff in the Steam Generator branch of NRR, in accordance with the guidance in Section 03.04, “Steam Generator Tube Inspection Activities,” of IP 71111.08.

0609J-02 BACKGROUND

Historically, the first probabilistic risk assessments accounted only for the sequences initiated by spontaneous tube rupture events during normal operation. In the mid-1980s, NUREG-0844 (ML082400710) considered the pressure-induced ruptures in sequences initiated by steam-side depressurization (e.g., a main steam line break (MSLB)) causing one or more degraded SG tubes to rupture or in sequences caused by failure of the reactor protection system when feedwater is lost (LOFW-ATWS), and NUREG-1150 (ML010140729), published in 1990, considered the high-temperature-induced rupture sequences. In the mid-1990s, NUREG-1570 (ML070570094) collected all these sequences in one place and evaluated them for a specific level of degradation. In 2018, NUREG-2195 (ML16134A029) summarized consequential SG tube rupture (C-SGTR) analyses of replacement SGs with thermally treated Alloy 600 and Alloy 690 tubes. This SDP incorporates information obtained from the NUREGs and available industry information to provide a generic guidance for assigning a preliminary significance to inspection findings when SG tube degradation has violated one or more tube integrity performance criteria. For more information regarding the core damage accident sequences and the technical basis of this SDP, refer to IMC 0308, Attachment 3, Appendix J, “Technical Basis for Steam Generator Tube Integrity Findings.”

0609J-03 GUIDANCE

This appendix places typical tube degradation inspection findings in broad significance groups, represented by Green, White, Yellow, or Red colors. According to the ROP, “Green” findings are those that result in a change to the large early release frequency (ΔLERF) below 10^{-7}/reactor-year. “White” findings are in the ΔLERF range between 10^{-7} and 10^{-6}/reactor-year. “Yellow” findings are in the ΔLERF range between 10^{-6} and 10^{-5}/reactor-year. “Red” findings are those with ΔLERF above 10^{-5}/reactor-year.

Table 1, Steam Generator Tube Integrity SDP Matrix, below, presents the information that is used to determine the preliminary significance of inspection findings. It is expected that region-based in-service inspection (ISI) inspectors who normally review licensee SG tube integrity test results will be the primary users of Table 1. Resident inspectors may use the guidance, but their assessment should be reviewed by the region-based ISI inspector. Using Table 1, any finding determined to be White, Yellow, or Red, or that includes performing a Detailed Risk Evaluation, must be reviewed by a senior reactor analyst or a senior risk analyst with experience in SG tube risk assessment. Assistance is also available from NRR/Division of Risk Assessment (DRA)/Probabilistic Risk Assessment Oversight Branch (APOB).

Table 1: Steam Generator Tube Integrity SDP Matrix

Preliminary Significance	ΔLERF/reactor-year	Degree of Tube Degradation Associated with Inspection Finding
RED[a]	$\Delta LERF > 10^{-5}$	Any condition that results in: Tube burst during normal operations Tube(s) found during testing to have been susceptible to burst during normal operations Tube(s) found during testing that could not sustain ΔP_{MSLB} (B&W)[c]
YELLOW[a]	$10^{-6} < \Delta LERF < 10^{-5}$	One tube that cannot sustain ΔP_{MSLB} (W and CE)
WHITE[a]	$10^{-7} < \Delta LERF < 10^{-6}$	One tube that cannot sustain $3x\Delta P_{NO}$ (W and CE)
GREEN[a]	$\Delta LERF < 10^{-7}$	One or more tubes that should have been plugged or repaired based on a previous inspection (refer to discussion in 0609J-04)
Perform a Detailed Risk Evaluation (based on parameter values specific to individual findings)	ΔLERF potentially $> 10^{-7}$	Two or more tubes that cannot sustain $3x\Delta P_{NO}$ One or more tubes that cannot sustain $3x\Delta P_{NO}$ in two of last three inspections One or more SGs that violate “accident leakage” performance criterion One tube that cannot sustain $3x\Delta P_{NO}$ (B&W)[b,c]

Notes:

[a] The assigned colors for Table 1 assume that the releases from core damage events with failed tubes have characteristics that are appropriately treated as part of the large early release frequency as modeled by the NRC in NUREG-1150.

[b] Babcock and Wilcox (B&W) SGs with circumferential tube cracks may be susceptible to failure due to axial stresses induced by thermal transients. If circumferential cracks are found in the free-span of a B&W plant, the issue should be submitted for a Detailed Risk Evaluation.

[c] B&W SGs are listed separately for some findings because they have different frequencies for some important sequences. High/dry core damage sequences are less likely to produce tube failures due to high tube temperatures in B&W once-through SG designs than in the U-tube SG designs in Westinghouse (W) and Combustion Engineering (CE) plants. Also, B&W plants have a higher incidence of steam-side depressurization events that would fail tubes that had degraded to the degree that they are susceptible to MSLB accident pressures.

0609J-04 DISCUSSION

Steam Generator Tube Degradation

Because tube degradation that violates the SIPC ($3\Delta P_{NO}$) may make the tube susceptible to high/dry core damage sequences that have a frequency in the low-10^{-5}/reactor-year range, any of the significance colors are possible. However, the degree of degradation beyond the performance criterion, the fraction of a year over which this degree of degradation existed, and many plant-specific factors are important determinants for the risk in a specific case. Information gathered through previous plant-specific analyses and engineering judgment have been used to assign a "White" significance level for findings of single tubes that are susceptible only to these sequences. When multiple tubes have degraded below the SIPC, or a single tube has degraded below that level in multiple cycles, it is more likely that the total risk will fall into the "Yellow" range. For that reason, Table 1 indicates "perform a Detailed Risk Evaluation" for findings involving multiple instances of exceeding the SIPC. The B&W SGs with one tube that violates the SIPC are also listed under the "perform a Detailed Risk Evaluation" category because the lesser degree of susceptibility for the once-through design to the high/dry sequences provides a substantial potential for a "Green" result.

When one or more tubes have degraded to the point that they cannot sustain the maximum pressure differential expected during a design-basis main steam line break event (ΔP_{MSLB}), it is also necessary to include those sequences in the risk assessment. The threshold for these sequences is the lowest operable pressurizer valve setpoint. In some plants that will be the pressurizer power-operated relief valve (PORV); for other plants where the PORVs are blocked or not installed, it will be the pressurizer safety relief valve setpoint. Again, B&W plants differ significantly from the W and CE plants and B&W plants have experienced several events that produced pressures near these thresholds shortly after a reactor trip. Westinghouse plants have experienced a relatively smaller number of events (considering the numbers of each design in operation), and none of these events the staff is currently aware of produced such large pressure differentials across the tubes after a reactor tripped from normal operation.

However, Westinghouse plant events are known to have produced similarly large pressure differentials across the tubes under other operational situations and smaller pressure differentials following trips from full power. On this basis, the assumed frequency of a steam-side depressurization event is estimated at about 10^{-2}/reactor-year for B&W plants and about 10^{-3}/reactor-year for the plants with U-tube designed SGs. When degradation has made the tubes susceptible to rupture if a SG depressurizes, a depressurization event becomes much more difficult for the operators to handle. As noted in section 7.4.4 of NUREG-2195, when considering the difficulty of the combined primary and secondary system failures, the probability of the plant operators failing to diagnose the occurrence of a C-SGTR after the steam line break scenario was estimated using the Standardized Plant Analysis Risk Human Reliability Analysis (SPAR-H) worksheet to be about 2.5×10^{-2}. Therefore, a tube in a Westinghouse SG that was susceptible to a C-SGTR from a steam-side depressurization was estimated to produce a $\Delta CDF/\Delta LERF$ in the range of $<1 \times 10^{-9}$ to 3.4×10^{-7}. For a CE SG, a tube that was susceptible to a C-SGTR from a steam-side depressurization was estimated to produce a $\Delta CDF/\Delta LERF$ in the range of $<1 \times 10^{-9}$ to 5.3×10^{-7}.

Finally, a performance deficiency that results in the amount of degradation that makes a plant susceptible to tube rupture during normal operation has been assigned a "Red" significance for all plant designs. Included in this significance are tubes that would rupture at pressure differentials that are often encountered during normal plant operations, even if the tube did not actually rupture because the actual operations did not happen to include those pressures while the tube was susceptible. A probability of about 0.1 for encountering those pressures is sufficient to keep the ΔLERF estimate in the "Red" category. The pressure threshold for this category is about 1600 psi for many plants. However, some plants may subject their tubes to much higher values, so plant-specific information should be used.

This appendix includes a "Green" criterion for plant operation at-power with one or more tubes that should have been repaired or plugged but were not. This criterion is intended to apply to either 1) a licensee's failure to identify a flaw that should have been identified as meeting the plugging limit with the data obtained in a previous inspection, or 2) a licensee's inadvertent failure to plug a tube that was identified for plugging. This criterion does not apply to the situation where a tube that is identified as flawed in a subsequent inspection can be found to have exhibited a detectable signal in the previous inspection data, unless the data from the previous inspection clearly indicates that the flaw exceeded the plugging limits at the time of the previous inspection. However, if the flaw causes the tube to fail the $3x\Delta P_{NO}$ requirement when it is found in the subsequent inspection, then SDP criteria listed under White, Yellow, or Red will still apply.

Findings involving accident leakage have been placed in the "perform a Detailed Risk Evaluation" category of Table 1 because the wide range of potential leak rates can result in risk levels that range from the "Green" into the "Red" categories. Individual findings that involve degradation that would exceed the AILPC under design-basis accident conditions should be referred to a regional senior risk analyst with assistance from NRR/DRA/APOB. The analyst will compare the finding parameters to the latest information available from the ongoing research efforts to select an appropriate significance for the Detailed Risk Evaluation analysis.

Table 1 does not include entries for exceeding the operational leakage limits because that does not necessarily mean that a significant risk increase has occurred. When that limit is exceeded, the licensee must shut down the plant and find the cause. Once the cause is determined, it will be possible to characterize the problem in terms of the probability for rupture and the estimated rate of leakage at the specific conditions associated with the risk significant accident sequences. Inspectors are advised to make sure they have all the pertinent information before going into the process of identifying violations and/or performance deficiencies and assessing their significance. Therefore, the significance can then be based on the entries for those findings in Table 1. For more information regarding the operational and accident-induced leakage criterion, refer to IMC 0308, Attachment 3, Appendix J, "Technical Basis for Steam Generator Tube Integrity Findings."

The B&W reactors have an additional issue that is not relevant to the U-tube designs used by Westinghouse and CE. The B&W design uses straight tubes that can be put into tension or compression by thermal transients in the reactor coolant system, due to changes in the temperature difference between the tubes and the SG vessel shells, which are rigidly connected, parallel mechanical structures. For transients that cool the tubes significantly more rapidly than the shells, the tubes may experience axial tension loads that are high enough to cause tube failure at significant circumferential cracks. At present, significant circumferential cracking is not being found in the free span at B&W plants. If it is found, it should be carefully evaluated for the thermal loads as well as the pressure loads. The SDP does not attempt to assign significance to a finding of significant circumferential cracking in the free span of the tubes in B&W reactors, but it does include a note to alert inspectors to submit the finding for the performance of a Detailed Risk Evaluation if it ever occurs.

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0609J-05 REFERENCES

ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” and Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components,” The American Society of Mechanical Engineers, [various editions]

IMC 0308, Attachment 3, Appendix J, “Technical Basis for Steam Generator Tube Integrity Findings”

IP 71111.08, “Inservice Inspection Activities”

NUREG-0844, “NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity” (ML082400710), September 1988

NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants” (ML040140729), December 1990

NUREG-1570, “Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture” (ML070570094), March 1998

NUREG-1740, “Voltage-Based Alternative Repair Criteria” (ML010750315), March 2001

NUREG-2195, “Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes” (ML18122A012), May 2018

END

Attachment 1: Revision History – IMC 0609, Appendix J

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description of Training Required and Completion Date	Comment Resolution and Closed Feedback Form Accession Number (Pre-Decisional, Non-Public Information)
N/A	05/06/04 CN 04-010	Revision History reviewed for last four years. IMC 0609 SDP Appendix J has been created to address Steam Generator Tube Integrity	N/A	N/A
N/A	ML102500252 07/06/11 CN 11-011	Revision removes guidance to open an unresolved item for ISI Programmatic findings and changes issues that were “to be determined” to “assess in Phase 3.” The Phase 1 portion of the appendix was moved to IMC 0609.04 Phase 1 Screening (ROPFF 0609J-1356).	N/A	N/A
	ML20169A502 10/16/20 CN 20-051	Five-year periodic review completed. Revised document based on organizational changes at the NRC and changes made to other IMC documents. Updated formatting based on IMC 0040.	N/A	ML20169A501
	ML26084A623 04/30/26 CN 26-016	Updated for consistency with IMC 0308 Attachment 3 Appendix J – Technical Basis for SG Tube Integrity Findings, which was updated in February 2024. Updated formatting based on IMC 0040. Removed references to Phase 1, 2, and 3 as the terms are being phased out. Changes made for consistency between IMCs 0609 App A, 0609 App J, 0308 Att 3 App J, and 0327. These revisions were recommended as a result of the ADVANCE Act 507 Report to Congress that discussed the revision of the ROP Baseline Inspection Program.		ML25274A088 ML25301A230 ML24051A092 0609J-2428 ML21335A424