**APPENDIX J**

**STEAM GENERATOR TUBE INTEGRITY FINDINGS SIGNIFICANCE DETERMINATION PROCESS**

## 1.0 INTRODUCTION

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

The SDP is not suitable for assigning significance to findings that involve only programmatic deficiencies in the licensee's steam generator tube integrity inspection program without knowing the consequence of those deficiencies on actual physical tube integrity. If a programmatic deficiency is identified during the inspection, the inspector shall notify NRC headquarters technical staff (NRR/Division of Component Integrity (DCI)) in accordance with the guidance in Section 03.04 – Steam Generator (SG) Inspection Activities of IP71111.08.

This SDP is entered from the Phase 1 SDP (Attachment 0609.04) as a result of a degraded SG tube condition involving at least one tube that cannot sustain 3 times the differential pressure across a tube during normal full power, steady state operation ($3\Delta P_{NO}$) or one or more SGs that violate "accident leakage" performance criterion (i.e., involve degradation that would exceed the accident leakage performance criterion under design basis accident conditions). The Phase 2 SDP provides generic guidance for assigning the preliminary "color" to inspection findings when steam generator tube degradation has exceeded tube integrity performance criteria. Table 1, Steam Generator Tube Integrity SDP Matrix, found in Section 3, presents the guidance for determining the preliminary significance of steam generator tube integrity findings.

## 2.0 BACKGROUND

Typical PRAs account only for the sequences initiated by spontaneous tube rupture events during normal operation. In the mid-1980s, NUREG-0844 identified the pressure-induced ruptures in sequences initiated by steam-side depressurization 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 identified the high-temperature-induced rupture sequences. In the mid-1990s, NUREG-1570 collected all of these sequences in one place and evaluated them for a specific level of degradation. A few plant-specific PRAs have been updated to incorporate the induced-rupture sequences. This SDP incorporates information obtained from the NUREGs and available industry information to provide a generic guidance for assigning a preliminary "color" to inspection findings when 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 308, "Reactor Oversight Process Basis Document."

## 3.0 GUIDANCE

This appendix places typical tube degradation inspection findings in broad "color" groups. According to the ROP, "Green" findings are those that result in a 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 ISI inspectors who normally review licensee steam generator 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 conditions to be assessed in Phase 3, must be reviewed by a senior reactor analyst or a risk analyst with experience in steam generator tube risk assessment. Risk analysts who have this expertise are in the Division of Risk Assessment of NRR.

**Table 1**
**Steam Generator Tube Integrity SDP Matrix**

| **Preliminary Color** | **$\Delta$LERF/reactor-year** | **Degree of Tube Degradation Associated with Inspection Finding** |
|---|---|---|
| RED | $\Delta$LERF > $10^{-5}$ | Any condition that results in:<br><br>Tube burst during normal operations<br><br>Tube(s) found during testing to have been susceptible to burst during normal operations<br><br>Tube(s) found during testing that could not sustain $\Delta P_{MSLB.}$ **(B&W)** |
| YELLOW | $10^{-6}$ < $\Delta$LERF < $10^{-5}$ | One tube that cannot sustain $\Delta P_{MSLB}$ **(W and CE)** |
| WHITE | $10^{-7}$ < $\Delta$LERF < $10^{-6}$ | One tube that cannot sustain 3x$\Delta P_{NO}$ **(W and CE)** |
| GREEN | $\Delta$LERF < $10^{-7}$ | One or more tubes that should have been repaired as a result of previous inspection (refer to discussion). |
| Conditions to be assessed in Phase 3 (based on parameter values specific to individual findings) | $\Delta$LERF potentially > $10^{-7}$ | Two or more tubes that cannot sustain 3x$\Delta P_{NO}$<br><br>One or more tubes that cannot sustain 3x$\Delta P_{NO}$ in two of last three inspections<br><br>One or more SGs that violate “accident leakage” performance criterion<br><br>One tube that does cannot sustain 3x$\Delta P_{NO}$ **(B&W)** |

Notes: The assigned colors for Phase 2 are based on the assumption 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&W plants 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 Phase 3 analysis.

## 4.0 DISCUSSION

Babcock and Wilcox Reactors.

Babcock and Wilcox (B&W) reactors 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.

Steam Generator Tube Degradation.

Because tube degradation that violates the structural integrity performance criterion (typically 3 times the differential pressure across a tube during normal full power, steady state operation, $3\Delta P_{NO}$) 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 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 structural integrity performance criteria, or a single tube has degraded below that level in multiple cycles, it is more likely but not certain that the total risk will fall into the "Yellow" range. For that reason, Table 1 indicates "assess in Phase 3" for findings involving multiple instances of exceeding the structural integrity criteria. B&W plants with one tube that violates the structural integrity criteria are also listed under the "assess in Phase 3" 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 ($\Delta P_{MSLB}$), 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. 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 the staff is currently aware of that produced such high pressure differentials across the tubes after a reactor tripped from normal operation. However, Westinghouse plant events are known to have produced similarly high pressure differentials across the tubes under other operational situations and lesser 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 U-tube designs. When degradation has made the tubes susceptible to rupture if a steam generator depressurizes, a depressurization event becomes much more difficult for operator response. Considering the difficulty of the combined primary and secondary system failures, the probability for the plant operators failing to stop the sequence before core damage occurs is estimated to be about $10^{-2}$. Thus, a tube susceptible to steam-side depressurization event for a year is estimated to produce a ΔCDF and a ΔLERF of about $10^{-4}$/reactor-year for a B&W plant and about $10^{-5}$/reactor-year for a Westinghouse or Combustion Engineering plant. These values are well into the "Red" range for B&W plants and at the Yellow/Red threshold for the U-tube plants. Since susceptibility is not expected to occur for an entire year in most cases, the U-tube plants have been assigned a preliminary "Yellow" while the B&W plants are assigned a preliminary "Red."

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" color for all plant designs. Included in this color 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 "assess in Phase 3" 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 accident leakage performance criterion under design basis accident conditions should be referred to a risk analyst with expertise in steam generator risk assessments. The analyst will compare the finding parameters to the latest information available from the ongoing research efforts to select an appropriate color for the Phase 2 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. Therefore, the significance can then be based on the entries for those findings in the table. For more information regarding the operational and accident-induced leakage criterion refer to IMC 308, "Reactor Oversight Process Basis Document."

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 RCS, 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 a color 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 Phase 3 analysis if it ever occurs.

## 5.0 REFERENCES


1. NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity, NUREG-0844, U. S. Nuclear Regulatory Commission, September, 1988.

2. Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants, NUREG-1150, U. S. Nuclear Regulatory Commission, December, 1990.

3. Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture, NUREG-1570, U. S. Nuclear Regulatory Commission, March, 1998.

4. 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].

5. IMC 308, "Reactor Oversight Process Basis Document"

6. 71111.08 "Inservice Inspection Activities"


END

ATTACHMENT 1
Revision History – IMC 0609, Appendix J

| Commitment Tracking Number | Issue Date | Description of Change | Training Needed | Training Completion Date | Comment Resolution Accession Number |
|---|---|---|---|---|---|
| N/A | 05/06/04<br>CN 04-010 | Revision History reviewed for last four years.<br>IMC0609 SDP Appendix J was been created to address Steam Generator Tube Integrity | NO | N/A | N/A |
| N/A | ML102500252<br>07/06/11<br>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 IMC0609.04 Phase 1 Screening (ROPFF 0609J-1356). | NO | N/A | ML |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |