Appendix J

STEAM GENERATOR TUBE INTEGRITY FINDINGS SIGNIFICANCE DETERMINATION PROCESS

1.0 INTRODUCTION

The significance determination process (SDP) provides a method to place inspection findings in context for risk significance in a manner that allows them to be combined with other plant performance results. This information is used to determine the level of NRC engagement in accordance with the Reactor Oversight Process. This appendix is used in conjunction with Inspection Procedure (IP) 71111.08 “In-service Inspection,” to estimate the risk significance of steam generator tube integrity issues which 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 a 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 that is greater than minor significance, the inspector shall notify NRC headquarters technical staff (NRR/DE and/or NRR/DSSA) via regional management to consider risk significance and any immediate plant safety consequences. If the analysis allows for waiting until the next testing interval (e.g., refueling outage) to determine the significance of the programmatic deficiency, an unresolved item (URI) may be issued and documented in an inspection report. This URI will not be factored into the SDP timeliness metric.

This SDP, which accounts for Phases 1 and 2, provides generic guidance for assigning the preliminary “color” to inspection findings when steam generator tube degradation has exceeded tube integrity performance criteria. All inspection issues related to steam generator tube degradation must be initially screened for significance (reference Appendix B to IMC 0612) once a licensee performance deficiency is identified.

Plant-specific and degradation-specific factors can have substantial effects on the level of risk associated with individual findings. Where specific levels of degradation are most likely to be within the risk ranges associated with particular “colors,” this SDP indicates those colors. For a few types of findings that can have a wide range of physical parameter variation, this SDP indicates only “To Be Determined” because the range of the risk associated with the range of the determining parameter is much broader than the range of one color. 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

Because most probabilistic risk assessments (PRAs) contain only the logic for risk due to spontaneous tube rupture events, there is not yet a widespread recognition of the risk impact that results from lesser levels of tube degradation. Therefore, it has been determined that a full assessment of risk due to steam generator (SG) tube degradation requires consideration of several types of core damage accident sequences:

- Sequences initiated by the spontaneous rupture of a tube. The sequences that result in core damage involve a variety of combinations of equipment failures and human mistakes. Most of the core damage sequences also result in containment bypass.

- Sequences initiated by steam-side depressurization of a SG, which causes one or more degraded[1] tubes to rupture. These sequences result in core damage by similar combinations of equipment failures and human mistakes. Containment bypass is usually caused by the combination of tube rupture and the cause of the steam-side depressurization.

- Sequences created by initiating events and equipment failures that have nothing to do with the SG tubes. The core damage sequences of concern are characterized by relatively high reactor coolant system pressure and dry SGs at the time that fuel cladding oxidation occurs in the reactor core. These conditions subject the SG tubes to temperatures well above design values. At these abnormal temperatures, the tube material is weaker, and tube ruptures may occur if the tube strength has been degraded during normal operation. The effect of tube degradation on these sequences is an increase in the probability that containment bypass will occur for accidents already included in the base core damage frequency. They do not increase the core damage frequency.

- Sequences caused by failure of the Reactor Protection System to stop the nuclear chain reaction when feed water is lost. These sequences are called loss of feedwater anticipated transients without scram (lofw-ATWS) events. With additional equipment failures, they can produce reactor coolant system pressures that are high enough to cause other failures that lead to core damage. If the tubes are degraded, the high pressure may also rupture some tubes as well, creating a containment bypass.

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 the second and fourth types of sequences, and NUREG-1150 identified the high-temperature-induced ruptures in the third class of 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

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[1] In the context of this Appendix, the term “degraded” refers to any reduction in the structural/leakage integrity of a tube, regardless of the depth of the flaw. It is not intended to convey the special definition of a “degraded” tube used in the standard Technical Specifications.

tube integrity performance criteria. For more information regarding 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 assessed to be “To Be Determined” must be reviewed by a risk analyst with experience in steam generator tube risk assessment. Analysts who have this expertise are in the Probabilistic Safety Assessment Branch of NRR. Findings determined to be “Green” do not need to be reviewed further by a risk analyst.

Table 1
Steam Generator Tube Integrity SDP Matrix

Preliminary Color	ΔLERF/reactor-year	Degree of Tube Degradation Associated with Inspection Finding
RED	$\Delta LERF > 10^{-5}$	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 $\Delta P_{MSLB.}$ **(B&W)**
YELLOW	$10^{-6} < \Delta LERF < 10^{-5}$	One tube that cannot sustain ΔP_{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.
TO BE DETERMINED (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 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 (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.

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 judgement 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 only “To Be Determined” 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 “To Be Determined” 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 <u>operable</u> 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 "To Be Determined" 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.

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].

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