Methodology for Assessing Severe Accident-Induced Steam Generator Tube Rupture

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Executive Summary

Steam generator tube ruptures (SGTRs) are important to consider in nuclear plant risk assessments because radionuclides released from the primary system through the ruptured tube(s) could escape to the environment through openings in the secondary system and bypass the containment building. Most previous risk assessments have addressed SGTRs that occur during normal operation or during an accident, but prior to core damage. Very few risk assessments have considered SGTRs that occur after core damage. This has largely been due to a limited understanding of the phenomena that govern these severe accident-induced SGTRs (SAI-SGTRs).

This report outlines a methodology for assessing the frequency of SAI-SGTRs. The methodology has been developed based on the assumption that such an assessment would be a risk-informed application under USNRC Regulatory Guide 1.174. This assumption implies specific requirements for both the supporting probabilistic risk assessment (PRA) and the supplemental calculations that must be performed to support quantification of the SAI-SGTR frequency. The required PRA capabilities are outlined in this report along with a discussion of supplemental analyses that will be needed. The report also includes a recommended approach for calculating the conditional probability of SAI-SGTR given a specific accident scenario.

It should be noted that the methodology outlined in this report has not yet been applied to a sample plant. It is likely that, once the sample applications have been completed, aspects of the methodology will change. The methodology will be updated at that time and this report reissued. The updated report will include the sample applications and plant-specific insights based on the initial applications of the methodology.

It should also be noted that the USNRC is currently sponsoring work to obtain a better understanding of both SAI-SGTR phenomena and phenomena related to failures of other RCS components, such as the hot leg and surge line, that would preclude SAI-SGTR. The results from these ongoing studies could affect the proposed methodology and will be reflected in the updated report.

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1. Introduction

This report describes a recommended methodology for assessing the frequency of accidents resulting in severe accident-induced steam generator tube rupture (SAI-SGTR). The methodology has been developed based on the assumption that such an assessment would be a risk-informed application under Regulatory Guide 1.174 (USNRC, 1998b). This assumption implies specific requirements for both the supporting probabilistic risk assessment (PRA) and the supplemental calculations that must be performed to support quantification of the SAI-SGTR frequency.

1.1 *Description of SAI-SGTR*

Steam generator tubes constitute a substantial fraction of the reactor coolant system pressure boundary in a pressurized water reactor (PWR). Failure of one or more of the steam generator tubes could provide a pathway for release to the environment through the secondary side of the steam generator. This release path would bypass the barrier provided by the containment. There would be some radionuclide retention within the steam generator, but a substantial fraction of the radionuclides released from the primary system would escape to the environment. In addition, this release is likely to occur relatively early in an accident, before emergency evacuation procedures could be effective.

SGTRs can occur during normal operation (i.e., a SGTR initiating event), after another initiator and before core damage (e.g., following a main steam line break), or after another initiator and after core damage. The last of these, which is often referred to as a SAI-SGTR, is the subject of this methodology report. For the purposes of this report, it is assumed that, until core damage occurs, the steam generator tubes are still intact.

Loss of structural integrity of the steam generator tubes could result from elevated tube temperatures and elevated differential pressures across the tubes. The temperatures and differential pressures required to cause tube rupture depend on the characteristics of any flaws that may exist in the tubes due to any of the postulated tube degradation mechanisms (e.g., axial or circumferential stress corrosion cracking or damage from loose parts). Consequently, the assessment of SAI-SGTR must consider the initial tube flaw characteristics, the pressure and temperature histories experienced by the tubes during an accident, and the response of the tubes to these pressures and temperatures.

1.2 *Safety Functions Affected*

Based on the conditions that contribute to SAI-SGTR, the following safety functions are affected.

- Secondary side pressure control (affects differential pressure across the tubes)

- Secondary side inventory control (affects differential pressure across the tubes, temperature of the tubes, and scrubbing of the release)

- Primary side pressure control (affects differential pressure across the tubes)
- Primary side inventory control (affects temperature across the tubes)

System failures or human actions that influence these safety functions are likely to have a significant influence on the likelihood of a SAI-SGTR.

1.3 *PRA Risk Metric and Scope*

The key risk metric for this assessment is the frequency of a large early release of radioactive material (LERF) due to containment bypass, specifically through the steam generators. The scope of the assessment is assumed to be limited is as follows:

- Only accidents beginning with the reactor at full power are considered.
- If a SGTR occurs prior to core damage, the subsequent occurrence of SAI-SGTR is irrelevant to the ultimate plant end state (i.e., the containment bypass already exists prior to core damage). Therefore, SGTR sequences in which the tube rupture is the initiating event or is induced by some mechanism prior to core damage [e.g., anticipated transient without scram (ATWS)-SGTR or main steam line break (MSLB)-SGTR] are excluded from consideration.[1]
- Exclude medium LOCAs and larger – Preliminary estimates indicate that medium and larger LOCAs result in depressurization of the primary system and a breach in the primary large enough that the primary cannot be re-pressurized. While SAI-SGTR induced primarily by temperature is one possible failure mode, it is likely that the absence of some meaningful primary-to-secondary pressure differential will prevent occurrence of the conditions required for SAI-SGTR.[2]

1.4 *Outline of Methodology*

The American Society of Mechanical Engineers (ASME) Standard RA-S-2002 (ASME, 2002) sets forth requirements for PRAs used to support risk-informed decisions. The standard provides a flowchart illustrating the steps to follow when using a PRA for a risk-informed application. The methodology outlined in this report is adapted from the flowchart.

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[1] It is also possible that these conditions (e.g., ATWS, MSLB) could occur and not result in a SGTR prior to core damage. In such cases, a SAI-SGTR could occur later in the scenario. However, these would likely be very low frequency accident sequences and would be unlikely to have a significant effect on the frequency of containment bypass due to SAI-SGTR.

[2] As with many of the conclusions of this report, the final determination of the validity of this conclusion must await an actual application of the methodology to one or more specific plants.

As outlined in the standard, the general steps to be followed are:

1. Describe the issue to be assessed.
2. Identify the safety functions affected by the issue.
3. Identify the PRA scope and risk metrics needed to assess the issue.
4. Determine the Capability Category needed for each part of the PRA to support the application.
5. Review the existing PRA to determine whether it meets the Capability Categories established in step 4.
6. Review the Supporting Requirements (SRs) identified in the standard to determine whether they are sufficient in scope and level of detail to support the application.
7. Review the existing PRA to determine whether it satisfies the SRs at the appropriate Capability Category.
8. If the scope of the PRA or the SRs is insufficient, supplementary analyses or requirements are identified.
9. Complete the required supplementary analyses.
10. Upgrade the PRA to meet the Capability Categories for all identified SRs and to incorporate all new analyses.
11. Requantify the PRA to determine the impact of any changes on the risk metrics chosen in step 3.
12. Provide risk input to the decision makers.[3]

Steps 1 through 3 listed above provide a complete specification of the issue and the scope of the PRA. These steps have been completed and were discussed in Sections 1.1 through 1.3 of this report. Steps 4-8 define the capability requirements of the PRA and determine whether the existing PRA meets these requirements. Since this report addresses only the methodology, and is not an actual application to a specific plant, it is only possible to address the definition of capability requirements. Steps 4, 6, and 8 deal with the capability requirements, and the results of these steps for SAI-SGTR are addressed in this report. Steps 5 and 7 require that a specific plant application be

[3] The risk information provided by application of this methodology will guide the applicants in ascertaining the necessity and timeliness of steam generator tube repairs. Risk would be explicitly delineated in terms of the condition of the tubes relative to the thermal-hydraulic conditions that could occur during postulated accidents, and would, by its nature, encompass other steam generator safety issues such as aging.

undertaken, and so will be addressed when a specific application is conducted. Steps 9 through 12 involve completion of any required upgrades to the PRA and subsequent requantification of risk, if required, to provide the necessary input to the risk-informed decision. Again, these steps can only be completed when a specific plant application be undertaken.

Section 1 of this methodology report defines the scope of the SAI-SGTR issue and risk metrics to be used (the results of Steps 1 through 3 of the application process). Section 2 discusses the capability requirements for a PRA to enable it to be used for assessment of SAI-SGTR (the results of steps 4 and 6 of the application process). Section 3 discusses supplementary analyses that are needed and outlines an approach to determining the frequency of a large early release due to SAI-SGTR for a hypothetical plant (the result of step 8 of the application process).

As one may imagine, there is a certain amount of difficulty involved in taking the risk-informed application process from the ASME standard and completing certain key steps out of sequence. For that reason, there is a much greater confidence that the results of steps 1-4 are “robust” than is the case for the results of steps 6 and 8. While this report reflects our best judgment as to the methodology requirements for a comprehensive risk-informed application for SAI-SGTR, we fully anticipate that changes will occur to this methodology once a plant-specific application is undertaken, since specific analyses can reveal details and effects that are not possible to foresee when dealing in the abstract.

2. PRA Capability Requirements

In order to meet the requirements for risk-informed application, the utility's PRA must meet certain standards. The specific capability requirements for SAI-SGTR risk assessment are established in this section based on the assumption that such assessments would be a specific risk-informed application under Regulatory Guide 1.174 (USNRC, 1998b). As such, ASME standard RA-S-2002 (ASME, 2002) can be used to define the requirements for a PRA to be used to support an assessment of SAI-SGTR. The ASME standard establishes the required capabilities of the PRA for risk-informed application and provides a framework for identifying the need for PRA enhancements or special studies. When reading this report, it is essential that the reader also have a copy of the ASME standard available to aid in understanding the capability requirements discussed below. Also, it is important to understand that the ASME standard is an evolving document and is currently undergoing changes as a result of the NRC review process. The current status of that process can be ascertained by obtaining a copy of the draft regulatory guide (USNRC, 2002a) that endorses the standard with comments and exceptions, and a letter from the USNRC to ASME (USNRC, 2003) that further documents the agreed upon changes to the standard.[4]

This section describes the Capability Category requirements for each aspect of the PRA for assessing SAI-SGTR. The ASME has established three Capability Categories for each supporting requirement in the standard. The three Capability Categories are numbered from I to III with category III requiring the most detailed and plant-specific assessment. Table 2-1, which is copied directly from Table 1.3-1 of the ASME standard, discusses the basis for each Capability Category.

Each subsection below describes the Capability Category requirements for PRA areas identified in the ASME standard. Where appropriate, specific insights are given down to the supporting requirement level. Note that Capability Category requirements for documentation of each area of the PRA are not specifically addressed. It is expected that the documentation for each part of the analysis will correspond to the Capability Category of the underlying analysis.

In establishing the Capability Category requirements, there is a predisposition that Capability Category III is needed for most PRA supporting requirements dealing directly with SAI-SGTR accident scenarios. The reason is that these scenarios have generally not been dealt with in sufficient detail in prior PRAs for there to be past insights that scenarios. However, the requirement for achieving Capability Category III for those supporting requirements would apply only to the specific parts of the PRA model that affect SAI-SGTR, not across the entire model. In the discussion of capability category requirements provided below, whenever Capability Category III is specified, there is an elaboration of what parts of the PRA model need to provide this capability.

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[4] While the proposed changes do not have a significant impact on the selection of the required capability categories for a PRA to support application to SAI-SGTR, they will have an effect on the assessment of whether the PRA used for this application meets the required capability categories.

Table 2-1 Bases for PRA Capability Categories [Reproduced from Table 1.3-1 in (ASME, 2002)]

CRITERIA	CAPABILITY CATEGORY I	CAPABILITY CATEGORY II	CAPABILITY CATEGORY III
1. Scope and level of detail: The degree to which the scope and level of detail of the plant design, operation and maintenance are modeled	Resolution and specificity sufficient to identify the relative importance of the contributors at the system or train level including associated human actions	Resolution and specificity sufficient to identify the relative importance of the dominant contributors at the component level including associated human actions, as necessary [see Note (1)]	Resolution and specificity sufficient to identify the relative importance of the contributors at the component level including associated human actions, as necessary [see Note (1)]
2. Plant-specificity: The degree to which plant-specific information is incorporated such that the as-built and as-operated plant is addressed	Use of generic data/models acceptable except for the need to account for the unique design and operational features of the plant	Use of plant-specific data/models for the dominant contributors	Use of plant-specific data/models for all contributors, where available
3. Realism: The degree to which realism is incorporated such that the expected response of the plant is addressed	Departures from realism will have moderate impact on the conclusions and risk insights as supported by good practices [see Note (2)]	Departures from realism will have small impact on the conclusions and risk insights as supported by good practices [see Note (2)]	Departures from realism will have negligible impact on the conclusions and risk insights as supported by good practices [see Note (2)]

NOTES:

(1) The definition for Capability Category II is not meant to imply that the resolution and specificity is to a level to identify every SSC and human action; only those necessary for the specific SR. Similarly for Capability Category III, it is not meant to imply that the resolution and specificity is to a level to identify every sub-component for every component.

(2) Differentiation from moderate (conservative or acknowledged, potential non-conservative), to small, to negligible is determined by the extent to which the impact on the conclusions and risk insights could affect a decision under consideration. This differentiation recognizes that the PRA would generally not be the sole input to a decision. A moderate impact implies that the impact (of the departure from realism) is of sufficient size that it is likely that a decision could be affected; a small impact implies that it is unlikely that a decision could be affected, and a negligible impact implies that a decision would not be affected.

2.1 *Initiating Events Analysis*

Capability Category I is sufficient for the initiating events analysis. The onset of core damage is sufficiently long after the initiating event that the use of a standard set of initiating events and generic data will be adequate to represent the risk associated with SAI-SGTR.

2.2 *Accident Sequence Analysis*

In general, Capability Category II is sufficient for the accident sequence analysis. However, Capability Category III is required for those sequences where it is necessary to differentiate between core damage sequences by the physical conditions existing in the primary and secondary at the time of core damage that could affect SAI-SGTR. For example, it is acceptable in most applications that the success criteria for feedwater supply simply be in terms of "enough flow to prevent core damage" versus "not enough flow to prevent core damage." In the case of SAI-SGTR, the amount of feedwater supply that is available when there is "not enough" is an important factor in SAI-SGTR. Therefore, the accident sequence analysis needs to specifically differentiate between "no flow" and "some flow" cases leading up to core damage. Specific Category III areas include:

- AS-A8: End states cannot be defined simply when a core damage state is reached. The analyst will need to further develop and refine end states based on secondary and primary inventory and pressure at onset of core damage.

- AS-B3: Particular attention must be paid to the creation of thermal-hydraulic conditions that could affect the steam generator tubes after core damage begins.

- AS-B6: Other key time phased dependencies that could affect SAI-SGTR include recovery of feedwater or reactor coolant injection after it is too late to prevent core damage or in amounts too little to prevent core damage. Post core damage recovery could create conditions that increase the likelihood of SAI-SGTR.

2.3 *Success Criteria and Other Engineering Calculations*

In general, Capability Category II is sufficient. However, since there is not sufficient basis to determine the impact on the SAI-SGTR risk results from a less than thorough plant-specific assessment of these sequences, Capability Category III is required in certain areas. Specific Category III areas include:

- SC-A4: Specific success criteria for those functions that mitigate the effects of a severe accident on the steam generator tubes need to be defined. Special attention is required to define success criteria that will keep the conditions in the tubes below any threshold levels where potential SGTR

becomes a concern. Thus, success criteria for tube integrity itself also need to be developed in detail.

- SC-A5: The mission time for the steam generator tubes to remain intact is a complex function of the time to failure of the tubes and the time to failure of other parts of the reactor coolant system (RCS) that would relieve the pressure inside the tubes. Both of these are time distributions, not specific times, and are likely to be highly plant specific. Plant and sequence specific mission times for the tubes should be developed.

- SC-B1: Until such time as sufficient analyses of the risk from SAI-SGTR sequences has been performed for a number of plants and generic insights can be drawn, best-estimate, plant specific models should be used.

- SC-B2: While it is anticipated that lack of detailed technical information and models will require certain use of expert judgment in the analysis of SAI-SGTR, this should be kept to the absolute minimum possible, and preferably be used only to supply uncertainty estimates on key parameters used in the analytical models.

- SC-B3: Analysis of conditions that could lead to SAI-SGTR may challenge the capabilities of existing thermal-hydraulic codes. Models may need to be refined or, at the very least, limitations of the models should be reflected in the success criteria that are developed and in the basic event probabilities that are ultimately assigned.

- SC-B4: The analyst should recognize that all computer codes have limitations. One way to understand these limitations and to assess the uncertainty in model predictions is to compare results calculated using two different computer codes or results calculated by the same code but for two different plants.

- SC-B6: Once again, since there is little if any basis to conclude that conservative or optimistic assumptions will not affect the conclusions about the significance of SAI-SGTR, such assumptions should not be used.

2.4 *Systems Analysis*

In general, Capability Category II is sufficient for the systems analysis, including analysis of those systems that affect the physical conditions that could lead to SAI-SGTR. There are, however, certain areas where special considerations require that Capability Category III be provided. Specific Capability Category III areas include:

- SY-A11: Variable success criteria for systems whose function serves to affect SAI-SGTR should be developed, based on the results of the success criteria analysis. Both time-dependence and the degree of failure (i.e., the effect of partial versus total failure) will have significant effects on SAI-SGTR.

- SY-A13: Where component leakage rates would have a significant impact on the likelihood of SAI-SGTR (e.g., would affect the primary or secondary pressure or inventory), a spectrum of realistic leakage rates should be considered.

- SY-A19: For those systems whose function serves to affect SAI-SGTR, the use of engineering analysis to support modeling of system conditions that cause a loss of desired function is essential. The use of an assumed failure probability of 1.0 should not be used.

- SY-A20: Careful, plant-specific and detailed consideration should be given to the possibility that systems or components could continue to operate beyond rated or design basis capabilities (e.g., under severe accident conditions). This is particularly important for those systems and components that, if they continue to operate, can increase the possibility of SAI-SGTR.

- SY-A23: SAI-SGTR occurs over a long timeframe, allowing substantial time to repair hardware faults. All credible repairs that could affect SAI-SGTR, both in a positive or negative fashion, should be included in the system models and justified.

- SY-B7: Special considerations on support systems need to be included. Timing of support system failure (e.g., late in the sequence, even after core damage begins) while not affecting the occurrence of core damage itself could affect the likelihood of SAI-SGTR. As with front-line system failure, partial failures of support systems could be irrelevant to the occurrence of core damage but could affect SAI-SGTR.

- SY-B8: Special attention should be given to plant-specific aspects of spatial and environmental hazards that could affect system operation (or functional success) in the post-core damage timeframe.

- SY-B10: Special consideration needs to be given to plant specific aspects of partial failures or slow degradation of support system performance that could affect front-line systems that have a key role in SAI-SGTR. Since degraded performance of primary and secondary boundary systems (RCP seals, pressurizer valves, secondary isolation valves, etc.) could alter the conditions under which SAI-SGTR could occur, there may be conditions of degraded performance of the associated support systems that could lead to this occurring.

- SY-B11: The effects of core damage could cause conditions that are beyond the capability of initiation and actuation systems to properly interpret and respond to the conditions. This could result in either failure to actuate or inadvertent actuation of systems, which should be incorporated into the model to the extent that they impact SAI-SGTR.

- SY-B15: The effects of core damage could cause conditions that are beyond the environmental qualifications of structures, systems, or components (SSCs). This could result in dependent failures of the SSCs, which should be incorporated into the model to the extent that they affect SAI-SGTR.

2.5 *Human Reliability Analysis*

The capability category requirements for human reliability analysis (HRA) depend on the type of human failure event being considered. In accordance with the ASME standard, both pre-initiator and post-initiator human failure events are considered.

For the pre-initiator HRA, it is expected that Capability Category I will be sufficient. The progression of the severe accident scenario occurs over an extended timeframe, and it is unlikely that errors that resulted in unavailability of systems prior to the occurrence of the initiating event will still have a significant relevance to the occurrence of SAI-SGTR some hours later.

For post-initiator HRA, Capability Category II is generally sufficient. However, there are certain areas where Capability III is required. Specific Category III areas include:

- HR-E1: Because extensive HRAs have been performed with the key focus on emergency operating procedures/abnormal operating procedures (EOPs/AOPs) and these EOPs/AOPs have a high level of plant-to-plant commonality regarding the error-forcing context and other factors affecting the resulting human failure events (HFEs), Capability Category II is sufficient. This is not true for HFEs that would relate to the use of severe accident management guidelines (SAMGs), and so the HRA with regard to such HFEs should be Capability Category III.

- HR-E2: Special considerations are required for actions to recover a failed function where recovery of that function subsequent to the onset of core damage could affect the occurrence of SAI-SGTR. In general, where a recovery action is intended to help prevent core damage, the sequence only considers that the action is not successfully taken by time “x”, corresponding to the maximum time available to prevent core damage. The analysis does not consider that continued efforts to recover the function could continue even after time “x” and that recovery within some specified period could increase the possibility of SAI-SGTR. Such actions need to be considered in the model.

- HR-E3: For those actions that are included by virtue of HR-E2 (above), detailed talk-throughs should be performed focusing on the cues that would cause the operator to abandon the recovery actions that could increase the possibility of SAI-SGTR.

- HR-E4: For those actions that are included by virtue of HR-E2 (above), simulator observations should be used to confirm the response of the

operator and to confirm the availability of cues required for the operator to abandon the recovery actions in a timely fashion.

- HR-G1: Detailed analysis should be performed for all human failure basic events that occur after the onset of core damage and that relate to functions that affect SAI-SGTR. Since there are no generic insights that can currently be used to determine which SAI-SGTR sequences may or may not be dominant, screening values should not be used.

- HR-G2: Detailed plant-specific estimates for failure in cognition for human failure basic events that occur after the onset of core damage and that relate to functions that impact SAI-SGTR should be developed.

- HR-G4: Detailed plant-specific thermal/hydraulic analyses or simulations should be used to establish the time frames available to complete actions relevant to SAI-SGTR. Special attention should be given to the time frames during which carrying out actions *intended* to prevent core damage would no longer actually prevent core damage but could increase the likelihood of SAI-SGTR.

- HR-G5: For all post-core damage actions relevant to SAI-SGTR, the required time to complete an action should be based on walk-throughs or simulator observations, and should include a “mock” Technical Support Center (TSC) to account for time delays and communication issues as well as the additional scenarios associated with the TSC.

- HR-G7: For all post-core damage actions relevant to SAI-SGTR, a detailed, plant-specific assessment of the degree of dependence that exists between multiple actions of this type in the same accident sequence or cutset should be performed. This should also include accounting for the influence of pre-core damage actions on these post-core damage actions.

- HR-H1: A broad spectrum of recovery actions specific to the design of the plant should be considered. Because of the long timeframe of the progression of SAI-SGTR, consideration of actions to repair systems and components should be included. The analysis should be as realistic as possible.

- HR-H2: Detailed specific procedures do not exist for response after the onset of core damage. Rather, the TSC exists for the purpose of developing response strategies under severe accident conditions. Therefore, the requirement that a procedure be available for the inclusion of operator recovery actions is not valid for response to prevent or mitigate SAI-SGTR. A detailed assessment of the conditions existing in each potential SAI-SGTR scenario should be conducted and credit should be given for any recovery action that could realistically be envisioned by the TSC.

- HR-H3: Commensurate with the requirements expressed in H1 and H2 above, the accounting for dependencies among human failure events (HFEs) for recovery and between these recovery HFEs and other HFEs in the scenario being assessed should be as realistically as possible.

2.6 *Data Analysis*

Within the ASME standard, data analysis refers to the use of directly applicable historical data to estimate equipment failures and unavailabilities modeled in the PRA, and not to estimate probabilities of phenomenological conditions evaluated by the use of physical models. Therefore, Capability Category I is generally sufficient for the data analysis. The risk associated with SAI-SGTR will be controlled to a large extent by the phenomenological aspects of the accident progression, which will have large uncertainties. The use of mostly generic component failure and unavailability data will not have a significant affect on the results, and so the effort required to improve this dated is not warranted, even for those components with direct impact on SAI-SGTR. However, there are certain very specific cases where more extensive data analysis is required.

The specific cases of concern are as follows:

- Temperature, pressure, and time-dependent failures of steam generator tubes under severe accident conditions.

- Temperature, pressure, and time-dependent failures of the hot leg and pressurizer surge line under severe accident conditions.

- Time-dependent degradation failures that result in leakage through reactor coolant pump (RCP) seals, power-operated relief valves (PORVs), safety/relief valve (S/RVs), and secondary valves with isolation functions resulting from exposure to severe accident conditions, losses/degradation of support systems (e.g., lost or degraded cooling to RCP seals), or a combination of the two.

- Partial failure resulting in leakage through PORVs, S/RVs, and secondary valves with isolation functions.

Only the last of these can be addressed in the traditional data analysis sense of collecting and analyzing failure data. The others require the use of probabilistically-based modeling of the failures. The data analysis requirements in the ASME standard do not clearly address cases where basic event probabilities are not established through the use of generic or plant-specific experience, but rather are established through modeling. Special studies using probabilistic modeling tools (e.g., mechanistic models combined with Monte Carlo simulation) are required to address these areas. These will be discussed further in Section 3.

For the last case above (partial failure), the data analysis requirements need to achieve Capability Category III in a number of areas, as follows:

- DA-A1: For the PORVs, S/RVs, and secondary isolation valves, it is necessary to define plant-specific partial failure modes for leakage past the valves that correspond to the leakage conditions that were determined in the success criteria analysis as having an effect on SAI-SGTR.

- DA-C1: The generic sources for parameter estimates should be only those that can allow differentiation between total and partial failure for leakage failure modes for PORVs, S/RVs and secondary isolation valves.

- DA-C10: For the PORVs, S/RVs, and secondary isolation valves, the surveillance test procedure should be reviewed to determine that it can differentiate between full and partial failure and, if so, that it results in documentation of the test results in such a manner that the extent of partial failure can be accurately “binned” into the appropriate partial failure mode.

- DA-D3: For the PORVs, S/RVs, and secondary isolation valves, detailed, plant specific means and uncertainties should be developed. It is not possible at this time to restrict this only to those valves that contribute measurably to SAI-SGTR since, until a number of plant-specific assessments are completed, there is no technical basis upon which to make that determination.

- DA-D7: For the PORVs, S/RVs, and secondary isolation valves particular attention needs to be given to determining whether the modifications to plant design or operating practice would change the estimate of the relative contribution of the various partial failure modes of concern.

2.7 *Internal Flooding*

Not required. While internal flooding sequences can eventually lead to SAI-SGTR, it is judged that these will not contribute significantly to the total and can be ignored.

2.8 *Quantification*

The Level 1 quantification can for the most part be Capability Category I. This is certainly true of those sequences where the potential for SAI-SGTR does not exist. These sequences can be placed into a single bin and quantified with a simple point estimate. This will give sufficient perspective as to the relative contribution of SAI-SGTR to total core damage frequency. There are a number of areas where it is necessary to achieve Capability Category III in the quantification. Insights into these Category III areas are provided below.

- QU-A2: Point estimate CDFs are acceptable for non SAI-SGTR sequences. For SAI-SGTR sequences, mean CDFs using full propagation of uncertainty distributions for the important events affecting SAI-SGTR should be

developed. While it is not possible at this time to specify comprehensively what the important events are, it is possible to specify in large measure what they are not. Uncertainty in component unavailability in general and human errors prior to core damage will be small compared to the modeling uncertainties for the elements discussed in section 2.6 of this report, and so propagation of uncertainties is not warranted. For the elements discussed in section 2.6 of this report and for anything that falls in a gray area, full propagation of uncertainties should be included.

- QU-A4: Commensurate with the Capability Category III requirements discussed in HR-H1, and HR-H2, extensive consideration of feasible recovery actions that can affect SAI-SGTR should be included in each core damage sequence with the potential for SAI-SGTR.

- QU-B6: Detailed, plant-specific accounting for system successes in a manner that would result in the most realistic estimation of the frequency of SAI-SGTR should be included.

- QU-B7: Detailed, plant-specific deletion of cutsets containing mutually exclusive events should be performed for all sequences with the potential for SAI-SGTR.

- QU-B9: Modules, sub-trees, or split fractions should not be used to facilitate quantification sequences with the potential for SAI-SGTR for those top events that occur after the onset of core damage.

- QU-C2: For sequences with multiple HFEs that have been included at Capability Category III in accordance with HR-G7, a detailed, realistic, plant-specific assessment of the degree of dependency and the combined probability taking into account the dependency should be performed.

- QU-E1: For the events discussed as Capability Category III in DA-D3, detailed, realistic, plant-specific characterization of parameter uncertainty should be performed.

- QU-E2: For the elements discussed in section 2.6 of this report, the sources of model uncertainty and the assumptions made or models adopted in response to those uncertainties should be identified in detail.

2.9 *LERF Analysis*

The focus of this application is to assess the risk associated with LERF from SAI-SGTR. This puts an interesting spin on the requirements for the LERF analysis. Since SAI-SGTR is a containment bypass scenario, it has little relationship with LERF from containment failure. As such, the aspects of LERF resulting from containment failure, and thus the sequences, systems, human actions and other parts of the PRA model related to such failures, can be Capability Category I without affecting the conclusions

regarding SAI-SGTR. The aspects of LERF specifically relating to SAI-SGTR need to be Capability Category III. Specific insights are as follows:

- LE-A (all): In grouping the core damage sequences into plant damage states, consideration must be given to the parameters that affect the stress placed on the steam generator tubes at the onset of core damage. Not only will plant damage states need to be separated by conditions of primary and secondary pressure, but by the pressure differential across the tubes, the extent of availability of primary and secondary inventory control systems (i.e., how much flow is available), the existence of forced or natural circulation, and the extent of primary system leakage.

- LE-B1: All potential SAI failure modes of the steam generator tubes should be included.

- LE-C2: All feasible operator actions, including repair of equipment, that could affect SAI-SGTR should be included. In addition to favorable effects on SAI-SGTR from taking actions consistent with the EOPs and SAMGs, other procedures, and the TSC, negative effects resulting from continuing to follow these procedures after they will no longer have the desired effect should also be included.

- LE-C3: All risk significant mitigation actions that can be taken by the operating staff to either decrease the likelihood of SAI-SGTR or to reduce the magnitude of release after SAI-SGTR should be included as branch points. Detailed, realistic technical justification should be provided for this inclusion.

- LE-C5: Capability Category III insights related to systems models are discussed in the section on system analysis.

- LE-C6: Capability Category III insights related to human actions are discussed in the section on human reliability analysis.

- LE-C7: Capability Category III insights related to accident sequences are discussed in the section on accident sequence analysis.

- LE-C10: SAI-SGTR containment bypass should be treated in a detailed, realistic manner. This could include taking credit for reducing the magnitude of release by scrubbing, or taking credit for limited tube leakage at certain flaw locations (e.g., at locations in which the tube opening would be constrained by supporting structures such as tube support plates) or for certain flaw sizes. Any credit taken must be supported by detailed, realistic analyses.

- LE-D4: Realistic analysis of secondary side isolation capability following the occurrence of SAI-SGTR should be performed.

- LE-D5: A realistic analysis of the structural integrity of the steam generator tubes when subjected to the spectrum of post-core damage conditions should be performed. In addition, in order for the overall assessment of SAI-SGTR to be realistic, a realistic analysis should be performed of the structural integrity of other parts of the RCS that could prevent tube rupture if they failed first.

- LE-D6: Since secondary pressure affects the likelihood of SAI-SGTR, realistic treatment of the parts of the containment isolation function that will affect secondary side pressure should be performed. Various realistic levels of partial success should be considered.

- LE-F2: Uncertainty distributions on all key parameters affecting SAI-SGTR should be performed. The distribution parameters should be highly plant-specific. Of special concern is the appropriate consideration of the uncertainty in the time-to-failure of the steam generator tubes versus the pressurizer surge line and the hot leg. It is likely that the overall mean probability of SAI-SGTR for each sequence will be highly dependent on the overlap of these curves, and so a formal convolution to determine that mean should be performed.

3. Recommended Approach for Assessing the Frequency of SAI-SGTR

As discussed in section 2, there are a number of supplemental analyses that will have to be performed to allow quantification of the LERF due to SAI-SGTR. The structured approach discussed below has been developed so that the analyst focuses only on factors that are likely to influence the frequency of LERF due to SAI-SGTR. This approach was developed, in part, based on recent experience with analysis of pressurized thermal shock (PTS) (USNRC, 2002b). It also reflects knowledge gained since the publication of the NUREG-1570 report, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture" (USNRC, 1998a).

The recommended approach includes the following steps:

- Identify plant-specific characteristics of steam generator tube flaws (locations, type, length, depth, etc.).
- Develop screening criteria for conditions leading to SAI-SGTR. These criteria will be plant-specific and will depend on the characteristics of steam generator flaws at the plant.[5]
- Develop prospective list of accident scenarios that could lead to these conditions.
- Build accident progression event trees and fault trees to quantify the frequency of conditions that could threaten steam generator tube integrity.
- Use approved thermal-hydraulic models to analyze these accident scenarios to determine conditions in the steam generators (average temperature and pressure) as a function of time. Surge line or hot leg failure models should be disabled in these analyses.
- Develop uncertainty distributions for tube pressure and temperature.
- Run Monte Carlo analyses to develop uncertainty distribution for the time of SAI-SGTR in the absence of other RCS failures, such as the surge line or hot leg.[6] It is assumed that any one of these other RCS failures would preclude SAI-SGTR.
- Develop uncertainty distributions for the failure times of these other RCS components.

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[5] As an alternative to plant-specific analyses, it would be possible to develop generic screening criteria that apply to broad categories of plants with similar steam generator designs and flaw characteristics.

[6] In previous analyses, such as NUREG-1570, only hot leg and surge line failure were addressed. Recently it has been postulated that other components, such as the steam generator manway, could fail before a steam generator tube.

- Determine the probability of SAI-SGTR before other RCS failures based on the uncertainty distributions developed in the two preceding steps.

These steps are discussed in greater detail in the following subsections.

3.1 Characterize Existing Tube Flaws

Steam generator tubes exhibit a variety of flaw types, including circumferential cracks at the top of the tube sheet, axial primary water stress corrosion cracks (PWSCC) at roll transitions, freespan cracks, axial outer diameter stress corrosion cracks (ODSCC) at tube support plates (TSPs), circumferential cracks at TSP dents, axial cracks due to intergranular attack/stress corrosion cracking (IGA/SCC) in sludge pile areas, and flaws due to damage caused by loose parts. Existing tube inspection procedures are designed to detect most flaws before they reach a condition that could lead to tube failure under full power operation. However, some flaws are difficult to detect due to their location. In addition, because of human error, it is always possible that significant flaws may go undetected.

Because the probability of a steam generator tube rupture depends on the characteristics of the existing tube flaws, it is important to accurately determine the type, location, size, and depth of the existing flaws. When doing this, the analyst should consider the time since the last tube inspection, the history of tube flaw growth at the plant, and the demonstrated effectiveness of tube inspection procedures at the plant.

Researchers at Argonne National Laboratory (ANL) and Dominion Engineering, Inc (DEI) have developed methods for estimating the number and size distributions for flaws of various types in steam generator tubes (Gorman, et al., 1998). These methods estimate the number of tubes with detectable flaws as a function of time, and the size distribution for defects in these tubes. The probability of detection (POD) during in-service inspections was considered when determining the distribution of flaw sizes. Sample distributions were provided for lightly affected, moderately affected, and severely affected plants. Distributions were provided for six defect types: circumferential stress corrosion cracking, circumferential ODSCC at TSPs, free span ODSCC, IGA/SCC in hot leg sludge piles, axial ODSCC at TSPs, and flaws due to loose parts.

Although plant-specific flaw distributions should be used in the risk analysis whenever possible, in the absence of sufficient data to develop such distributions, use of the hypothetical distributions may be acceptable. The distributions should be selected based on plant-specific factors such as the age of the steam generators and past experience with steam generator degradation.

3.2 Develop Screening Criteria for the Conditions Needed for SAI-SGTR

A conservative set of screening criteria will be developed to characterize the pressure and temperature conditions that could lead to SAI-SGTR. The recommended approach for establishing these criteria include:

- Identify anticipated worst-case tube flaw characteristics for both axial and circumferential tube flaws.
- Use these worst-case flaw characteristics in tube integrity models to identify pressure and temperature conditions that could lead to tube failure.

The worst-case flaw characteristics should be selected so that they could reasonably be expected to exist in a steam generator near the end of a cycle (i.e., just prior to tube inspection).

In applying the tube integrity models, the analyst should recognize the uncertainty in the models and use conservative lower bound parameter values (i.e., values that would lead to conservative pressure and temperature estimates for the conditions at tube failure). This ensures that conditions that could lead to SAI-SGTR are not screened prematurely.

3.3 Determine Accident Scenarios That Could Lead to SAI-SGTR

A list of accident scenarios that could lead to SAI-SGTR will be determined with the help of thermal-hydraulic analysis. The thermal-hydraulic analyses will determine the complete or partial system failures, and human actions that could lead to the pressure and temperature conditions identified in the screening analysis.

When developing the list of accident scenarios, the analyst should consider both complete or partial failures (e.g., partially stuck open PORV), human errors of omission or procedure-driven errors of commission, and changes in the state of a component as the accident progresses (e.g., reclosure of a stuck open valve). The analyst should also recognize that the timing of failures or human actions may be important. For example, the thermal-hydraulic analysis may show that a stuck open PORV before time “x” or a human action to depressurize the primary system before time “y” will prevent SAI-SGTR or delay it sufficiently that hot leg or surge line failure would nearly always occur first. These factors should be considered when developing success criteria for prevention of SAI-SGTR.

3.4 Build Accident Progression Event Tress and Fault Trees

Accident progression event trees and fault trees provide a framework for assessing the frequency of postulated accident scenarios. They provide an estimate of the frequency of primary and secondary system conditions that could challenge tube integrity and the overall frequency that a tube rupture will occur. They also provide a means for characterizing the progression of accident and the resulting evolution of challenges to tube integrity.

The entry point of the SAI-SGTR event trees is core damage (the termination point of the Level 1 PRA) with the plant in a condition that would not preclude a SAI-SGTR or for which SAI-SGTR is no longer relevant. An example of the former condition might be a large or medium LOCA in which the RCS has been depressurized such that SAI-SGTR

is not possible. An example of the latter condition would be a tube rupture that occurs before core damage.

As noted in section 2, initial plant states important to SAI-SGTR may not be adequately captured in the Level 1 PRA. Thus, the analyst may have to do supplemental analyses to better define the initial plant states.

Once the initial plant states have been determined, the analyst must identify complete or partial failures of systems or components, or human actions that could lead to pressure and temperature conditions that could challenge tube integrity. These conditions become top events in the event tree. The branches of the event tree are then selected to correspond to the conditions of interest for SAI-SGTR.

To limit the number of accident sequences that must be developed and quantified, the thermal-hydraulic analyses should be used to group accident sequences based on similar pressure and temperature histories and, consequently, on similar likelihood of SAI-SGTR. For example, the thermal-hydraulic analyses may show that total RCP seal leakage greater than 400 gpm is sufficient to rapidly depressurize the primary system. If subsequent tube integrity analyses show that a depressurized primary system prevents SAI-SGTR for all probable tube flaws, then the event trees need only model accident sequences with RCP seal leakage less than 400 gpm. Accident sequence binning is likely to continue throughout the risk assessment process as new results from thermal-hydraulic and event tree analyses become available.

After the event trees and fault trees have been constructed, the next step is assignment of probabilities for basic events. As noted in section 2, there may be many instances in which the existing PRA may provide an inadequate basis for determining important basic event probabilities and supplementary analyses are required. Section 2 provides numerous examples of basic event probabilities for which supplementary analyses are likely to be needed due to the unique nature of SAI-SGTR events.

One of the strengths of the PRA framework is the ability to characterize uncertainties. When assigning basic event probabilities, the analyst should recognize the uncertainty in these probabilities and supply uncertainty distributions to the PRA model that adequately reflect the full range uncertainty. The aggregation of these uncertainties will be reflected in the overall uncertainty in the calculated SAI-SGTR frequency. The breadth of the uncertainty distribution is a measure of how robust the SAI-SGTR frequency estimates are.

3.5 Determine Pressure and Temperature Histories for Accident Scenarios

A number of computer codes for thermal-hydraulic analysis currently exist. In order to be used for analysis of SAI-SGTR, the candidate thermal-hydraulic codes must be approved by the USNRC specifically for this application. Currently, an approved list of thermal-hydraulic codes has not been established. It is expected that an approved list of codes will be established by the USNRC before the first risk-informed application of this methodology.

It is important that the thermal-hydraulic codes provide the user with the capability to vary uncertain model parameters in order to determine the impact of this uncertainty on the predicted conditions in the steam generator. For example, the analyst should be able to disable models for hot leg and surge line failure in order to isolate the calculation of SAI-SGTR from the effects of uncertainties in these models and in the conditions that cause these failures. The analyst should also be able to vary uncertain model parameters that could directly influence the calculated pressure and temperature in the steam generators. Modifications to the existing computer codes may be needed to provide the analyst with the necessary flexibility.

The analyst must also recognize that the thermal-hydraulic models are simplified representations of the real world. In some cases, these simplifications can directly affect the calculated pressure and temperature in the steam generator. In particular, the thermal-hydraulic codes utilize control volumes and heat structures and calculate average temperatures and pressures in these volumes and structures.[7] Use of an average temperature or pressure to predict failures that are inherently sensitive to pressure and temperature could result in significant prediction errors. In many cases, it cannot be determined *a priori* whether the errors are conservative or non-conservative. For example, calculations performed using detailed computational fluid dynamics (CFD) models have shown that averaging the temperature in the lower plenum of the steam generator can lead to a significant underprediction of the temperature in the hottest steam generator tubes. Underprediction of the tube temperature would lead to an artificial delay in the predicted time of tube failure for these hot tubes.

Once the model simplifications have been recognized by the analyst, it is possible to address the effects of these simplifications in the analysis. For example, the analyst could specify an uncertainty distribution for the tube temperature to account for non-uniform mixing in the lower plenum.

3.6 Develop Uncertainty Distributions for Pressure and Temperature Histories

Based on the thermal-hydraulic analyses discussed in the preceding section, the analyst will develop uncertainty distributions for the pressure and temperature histories in the steam generators. When developing these distributions, the analyst should consider all important uncertainties, including uncertainties in both model parameters and in the models themselves. With respect to model uncertainties, the analyst may consider calculated results using other approved thermal-hydraulic codes.

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[7] Most heat transfer models in the thermal-hydraulic codes divide heat structures into discrete nodes and calculate the temperature distribution in the heat structure based on this nodalization. The calculated nodal temperature is the average across the node. In heat structures with steep temperature gradients, the accuracy of this approach will depend on the nodalization chosen by the analyst.

3.7 Determine Uncertainty Distributions for the Time of SAI-SGTR

Previous studies of steam generator tube failure under severe accident conditions have shown that tube failure under severe accident conditions is governed by creep rupture phenomena. Majumdar, et al., (1998) provided creep rupture models for axial and circumferential cracks that compared well to data from a substantial number of tests at ANL. Their model considers the characteristics of the axial and circumferential flaws, the material properties of the tube material, and the time-dependent pressure and temperature experienced by the tubes.

In order to determine an uncertainty distribution for the time of tube failure, it is necessary to incorporate the creep rupture models into a probabilistic framework. If the creep rupture model is coded in spreadsheet form, spreadsheet add-in modules can be used to simplify creation of uncertainty distributions. Two of these add-in modules, Crystal Ball (a product of Decisioneering, Inc.) and @Risk (a product of Palisade Software, Inc.), have been widely used for similar applications, and a number of other commercial off-the-shelf software tools are available with capabilities similar to Crystal Ball and @Risk. In addition, stand-alone software tools may be developed and used as long as they have been adequately tested and validated.

When determining the uncertainty distributions for the time of SAI-SGTR, it is important to consider all important uncertainties in the inputs to the creep rupture models. These include uncertainty in the tube flaw characteristics, uncertainty in the creep rupture parameters, and uncertainty in the calculated pressure and temperature histories.

As noted previously, some tube failures may not produce tube openings that would result in high leakage from the RCS. The tube openings may be constrained by tube support plates or blocked by “crud” buildup. In addition, tests at ANL have shown that tubes with some initial flaw characteristics leak when they fail rather than rupture. Since we are interested only in large early releases from the containment, credit could be taken for cases in which leakage from the failed tube is limited. This could be done by assigning a conditional probability of high leakage (i.e., a “rupture”) given that the tube has failed. Since these effects are specific to certain flaw types and flaw locations, the conditional probability would be applied to the SAI-SGTR probability calculated for those flaw types and locations.

Uncertainty distributions for the time of SAI-SGTR would be developed for each tube degradation mode found at the plant. These individual distributions could be combined to form a single uncertainty distribution for the overall time of SAI-SGTR in the absence of other RCS failures (e.g., hot leg or surge line failure) that would preclude SAI-SGTR. Alternatively, the individual distributions could be kept separate and then combined later in the analysis (see section 3.9). When combining the individual distributions, the number of tubes exhibiting a given degradation mode must be considered.

3.8 Determine Uncertainty Distributions for the Time of Other RCS Failures

Steam generator tube rupture can only occur if there are no other significant failures of the RCS pressure boundary. In accident scenarios that have a steam generator at elevated temperature and pressure, the most likely failures are at the hot leg or surge line. Although the following discussion focuses specifically on hot leg and surge line failure, the analyst should recognize that other failures could occur and should consider these failures in the analysis.

As with failures of the steam generator tubes, failures of the hot leg and surge line under elevated temperature and pressure conditions are assumed to be by creep rupture. Because of uncertainties in the thermal-hydraulic conditions in the hot leg and surge line and uncertainties in creep rupture parameters, the times of hot leg and surge line failure are uncertain.

Thermal-hydraulic models such as RELAP5 often contain analytical models for hot leg and surge line failure based on creep rupture. It would be possible to use a thermal-hydraulic model to capture the uncertainty in the failure times if the model provides the user with adequate flexibility in varying model inputs. The alternative to this would be using a stand-alone creep rupture model that would accept uncertainty distributions for creep rupture parameters and the thermal-hydraulic conditions at the hot leg and surge line, and would calculate the resulting uncertainty distributions for the times of hot leg and surge line failure.

3.9 Calculate the Frequency of Large Early Release Due to SAI-SGTR

With probability distributions determined for the time of hot leg failure, surge line failure, and steam generator tube rupture, it is possible to determine the probability that a steam generator tube ruptures prior to either hot leg or surge line failure. This can be accomplished using standard probability analysis techniques or Monte Carlo analysis. The probability of SAI-SGTR would be determined for all modeled accident scenarios. Multiplying the SAI-SGTR probability by the frequency of the accident scenario gives the frequency of SAI-SGTR for the given scenario. Combining all scenario SAI-SGTR frequencies gives the total SAI-SGTR frequency.

It is anticipated that the risk information provided by application of this methodology will guide the applicants in determining the necessity and timeliness of steam generator tube repairs. Risk would be explicitly delineated in terms of the condition of the tubes relative to the thermal-hydraulic conditions that could occur during postulated accidents. This assessment of risk would, by its nature, encompass other steam generator safety issues such as aging.

4. References

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APPENDIX A

ACROYNMS/ABBREVIATIONS

ACRONYMS/ABBREVIATIONS

ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
CDF	computational fluid dynamics
DEI	Dominion Engineering, Inc.
EOP	emergency operating procedure
HFE	human failure event
HRA	human reliability analysis
IGA/SCC	intergranular attack/stress corrosion cracking
LERF	large early release frequency
LOCA	loss of coolant accident
MSLB	main steam line break
ODSCC	outer diameter stress corrosion cracking
POD	probability of detection
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PTS	pressurized thermal shock
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
RCP	reactor coolant pump
RCS	reactor coolant system
SAI-SGTR	severe accident-induced steam generator tube rupture
SAMG	severe accident management guideline
SGTR	steam generator tube rupture
SR	supporting requirement
S/RV	safety/relief valve
SSC	structure, system, or component
TSC	technical support center
TSP	tube support plate
USNRC	United States Nuclear Regulatory Commission