OFFICE OF NUCLEAR REGULATORY RESEARCH INFORMATION LETTER (RIL) 09-003 CONSEQUENTIAL STEAM GENERATOR TUBE RUPTURE WORK IN THE U.S. NUCLEAR REGULATORY COMMISSION'S OFFICE OF NUCLEAR REGULATORY RESEARCH

Background and Objectives

The deterministic licensing basis for pressurized water reactors (PWRs) generally includes consideration of spontaneous steam generator tube ruptures (SGTR) because these events represent both a breach of the reactor coolant system (RCS) pressure boundary and a containment bypass release path. Traditionally, SGTR events have been included in risk assessment studies as random initiating events. This approach focuses on the assessment of the potential for SGTR events to lead to core damage rather than for the potential of a core damage event to lead to an SGTR. Conversely, consequential SGTRs (C-SGTRs) refer to those SGTRs that may be caused as a result of another initiating event or a severe accident condition that leads to failure of SG tube(s). If a C-SGTR event occurs because of another initiating event (e.g., a very large steam-line break leading to high differential pressure across the SG tubes), it could increase the plant core damage frequency. If the postulated C-SGTR occurs because of high-temperature conditions caused by severe accident conditions associated with core damage, the resulting containment bypass could increase the plant's large early release frequency (LERF). Due, in part, to the potential significance of C-SGTR events, the American Society of Mechanical Engineers (ASME) probabilistic risk assessment (PRA) standard explicitly requires the consideration of consequential SGTR events.

The objective of this document is to summarize the key methods and results of the C-SGTR work (Reference 1) performed by the Office of Nuclear Regulatory Research (RES). This work supports closure of Item 3.5, "Develop Improved Methods for Assessing the Risk Associated with SG tubes Under Accident Conditions," of the U.S. Nuclear Regulatory Commission's (NRC's) Steam Generator Action Plan (SGAP) (References 2 and 7). This document also describes other thermal-hydraulic and materials work done by RES that is related to the C-SGTR issues (References 3 and 4) and describes how these work products interrelate to the risk assessment work. No new technical information is generated in this document.

In 2001, NRC consolidated numerous ongoing SG activities under the SGAP. Task 3.5 of the SGAP built upon previous work contained in NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," and expanded the work to include updated analyses for a wider range of plant-operating states and accident initiators.

Task 3.5 contained seven subtasks:

a. Development of an integrated framework for assessing the risk for the high-temperature/high-pressure accident scenarios of interest.

b. Issue report describing improved methods and appropriate treatment of uncertainty for identifying severe accident scenarios that lead to challenges of the RCS pressure boundary.

c. Develop logic framework for improved PRA models of the scenarios identified above, including the impact of operator actions.

d. Identify scenarios, calculate the frequency of containment bypass events at an example plant, make indicated method improvements, and document the improved methods and results.

e. Extend and document the methods and model logic to include Combustion Engineering (CE) plants.

f. Extend and document the methods and model logic to include consideration of external events as initiators and low power and shutdown as initial conditions.

g. Extend the improved methods and logic to include consideration of core-damage sequences initiated by secondary depressurization events (such as main steam line break [MSLB] design basis accident scenarios) that induce tube rupture.

Completion of SGAP Task 3.5 was supported by work conducted under SGAP Task 3.4 that was intended to develop a better understanding of reactor coolant system conditions and corresponding component behavior under severe accidents. In particular, SGAP Item 3.4.h[1] required a systematic assessment of alternate RCS pressure boundary locations that were vulnerable to failure during a severe accident. RES addressed SGAP Task 3.5 and related Task 3.4.h in the time period of 2002–2009 (References 1, 3, 4, and 6). Figure 1 below illustrates the flow of information among the various research efforts cited in this document.

The PRA report of Reference 1 provides the main technical basis for closure of SGAP Task 3.5. This report built on an earlier PRA study conducted by the Sandia National Laboratories (SNL) (Reference 6). The SNL PRA report focused on the station blackout (SBO) core-damage sequences because these events were deemed to be the most likely candidates of severe SG tube challenges. The SNL PRA study utilized preliminary, and generally more conservative, thermal-hydraulic and materials analyses from two other ongoing RES projects (References 3 and 4).

The thermal-hydraulic and materials projects examined two main issues:

1. Prediction of system-level thermal hydraulics under various severe accident conditions using SCDAP/RELAP5 models.

2. Behavior of PWR RCS components, other than SG tubes, under severe accident conditions.

The reports associated with the above thermal-hydraulic and materials research serve as the technical basis for closure of SGAP Item 3.4.h. Because of scheduling limitations, the SNL PRA report was finalized prior to completion of the thermal-hydraulic and materials studies. However, the final thermal-hydraulic and materials analyses, while they indicate greater margin

[1] SGAP Task 3.4.h, “Perform a Systematic Examination of the Alternate Failure Locations in the RCS that are Subject to Failure due to Severe Accident Conditions,” was intended to identify other reactor coolant pressure boundary locations that may fail either before or after the inception of a consequential SG tube failure. Other RCS pressure boundary failures can either prevent the occurrence of a consequential SGTR or mitigate the consequences of a C-SGTR event.

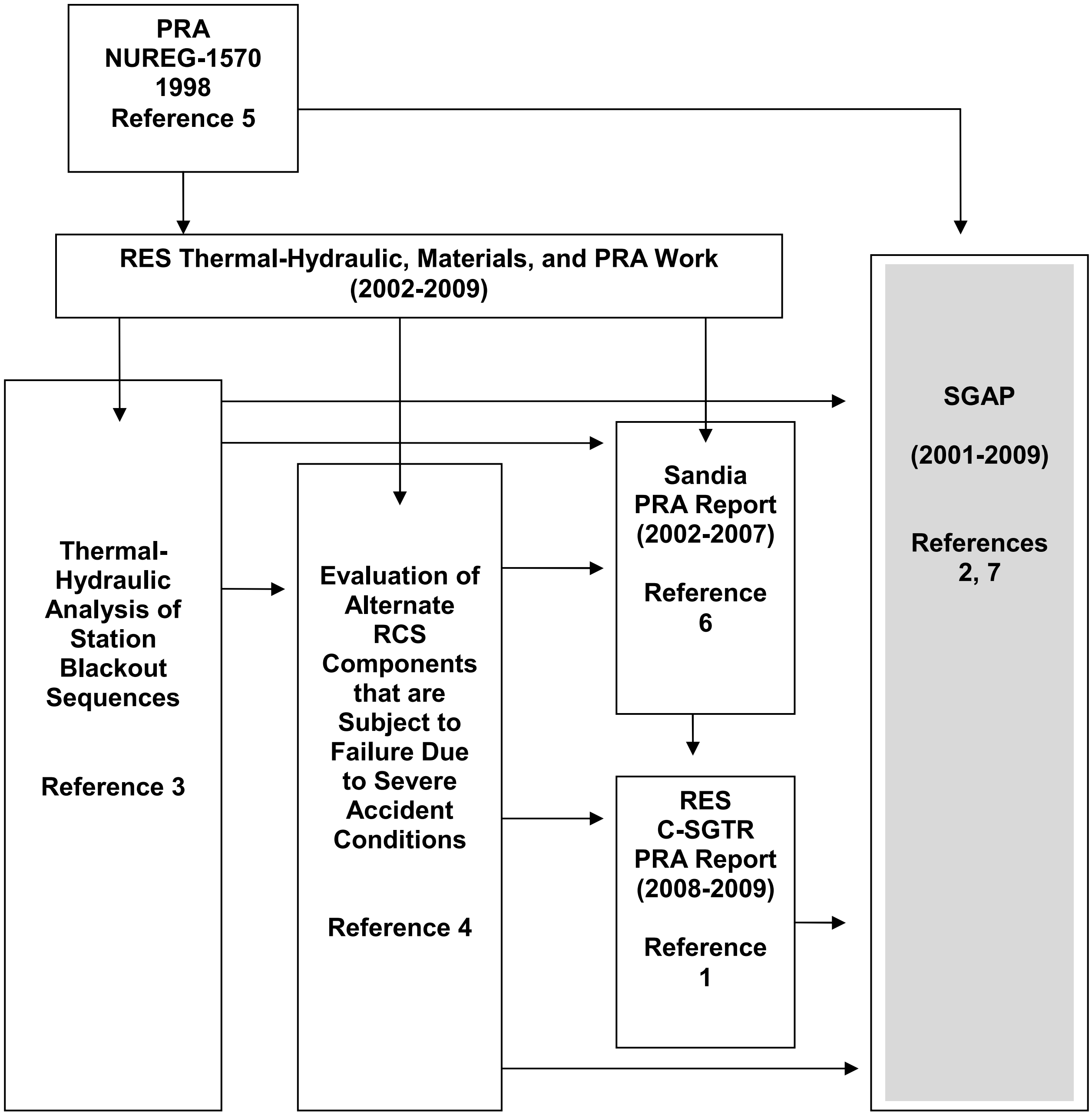

Figure 1. Flow Chart Illustrating the Relation of Documents and Work Mentioned in this Document

to consequential tube failure for Westinghouse plants, did not change the methodological approach for C-SGTR risk assessment described in the SNL PRA report. Planned future work in this area will use the final thermal-hydraulic and materials results to update the conditional failure probabilities for consequential SG tube failures.

The body of research described in References 1, 3, and 4 provided the basis for closure of all outstanding items assigned to RES in the SGAP. Resolution of the remaining issues associated with consequential SG tube failure risk no longer requires the level of coordination and agency focus required to implement the action plan process. Consequently, RES does not anticipate performing additional research activities under the SGAP. However, the closure of the SGAP does not preclude future consequential SG tube failure research activities. Future work activities associated with this topic will be coordinated using other agency tools such as the User Need and the Planning, Budgeting, and Performance Management (PBPM) processes.

Results

Both NRC and the nuclear industry have studied the potential C-SGTR events in detail for well over a decade. Based on this research effort, the conditions and sequences that can lead to C-SGTR are well studied and understood. However, previous studies have also demonstrated that C-SGTR risk is plant-specific and cannot be assessed on a generic basis. Moreover, the extensive PRA modeling changes needed to implement previous C-SGTR assessment methods limit the practicality of using these methods on a wide-scale basis. These conditions highlight the need for a C-SGTR assessment method that can accommodate plant-specific factors without the need for costly PRA model revisions.

In general, C-SGTR events fall into two categories: pressure-induced failures and temperature-induced failures. A pressure-induced tube failure can be caused by an increase in differential pressures across SG tubes when both the primary and secondary sides of the SG tubes are at normal temperatures. A failure of this nature could be caused from either secondary side depressurization or primary side overpressurization. Secondary side depressurization could occur from an MSLB or a transient with a stuck-open atmospheric dump valve. Primary side overpressurization can occur from large pressure excursions caused by an anticipated transient without scram (ATWS). A temperature-induced failure can be caused by the combination of high-differential pressure across the tubes and excessively high SG tube temperatures. These conditions commonly referred to as “high and dry”, are likely to occur during the core damage phase of certain severe accidents. The conditions for temperature-induced SG tube failures are achievable when the secondary side is dry (i.e., no auxiliary feed water is available) and an elevated primary-to-secondary system differential pressure is present. Events resulting in core damage with the RCS pressure near either the pressurizer power-operated relief valve (PORV) or safety relief valve (SRV) set points and with the secondary side dry and depressurized are generally considered to pose the greatest threat of temperature-induced SGTR. In these events secondary side depressurization is presumed to occur as a result of secondary side leakage. SBO event sequences account for the majority of the events that can result in these conditions. However, events where the RCS is at intermediate pressures (i.e., below normal operating pressure) and the SG’s secondary side is dry and depressurized may also pose a substantial threat to tube structural integrity. The RCS can be partially depressurized due to the failure of pressurizer PORVs or SRVs to close or reseat, or by a loss-of-coolant accident (LOCA) caused by the failure of reactor coolant pump

(RCP) seals or an alternate RCS pressure boundary location. The degree of depressurization depends on the timing and leak area associated with the RCS pressure boundary breach and with the accumulator injection set points.

The straightforward C-SGTR risk assessment method described in Reference 1 consists of the following main steps:

- Review of existing plant-specific PRA results to identify accident sequences that represent either a pressure or thermal challenge to SG tube integrity. In general, accident sequences can be classified into two broad categories: (1) severe accident sequences that thermally challenge SG tubes and (2) otherwise successful accident sequences (i.e., no core damage) that represent a significant pressure challenge to the SG tubes and may result in core damage due to a C-SGTR. Accident sequences that do not represent either a pressure or thermal challenge to the SG tubes are screened out from further consideration.

- Assessment of conditions that can mitigate the conditions challenging SG tube integrity. For example, operator actions to recover a source of secondary makeup to the SGs may reduce the potential for a thermally induced C-SGTR or actions to depressurize the primary system. These actions may include the incorporation of 10CFR50.44 (hh) measures (i.e., post 9/11 mitigation).

- Assessment of the conditional probability of a C-SGTR given the occurrence of accident sequences that challenge tube integrity and the failure of mitigative actions.

- Calculation of the incremental increase in containment bypass frequency based on the frequency of accident sequences that challenge SG tube integrity, the failure of mitigative actions, and the conditional probability of consequential tube failure.

This assessment method was illustrated through an example application to three PWR plant designs (two Westinghouse plants and one Combustion Engineering plant). For the purposes of illustration, the model parameters are based on previous thermal-hydraulic and PRA studies, as modified by expert judgment, and are believed to be realistically conservative. Although C-SGTR risk is plant-specific, RES believes that the illustrative conditional probability values used in the report provide a representative assessment of consequential SGTR risk. It should be noted that large uncertainties are associated with parameters used in this study and an accurate estimation of C-SGTR risk requires a detailed assessment of plant-specific model parameters.

Calculation of Conditional SG Tube Failure Probability

A key input for the C-SGTR risk assessment method is the conditional failure probability of the SG tubes during a severe accident. The SNL PRA report describes the methodology used to calculate this failure probability based on the following plant-specific inputs:

- Probability distributions for the length and depth of each flaw and the number of flaws.
- The time-dependent pressure difference and temperature experienced by the flaw.

- Probability distributions for the failure time of other RCS components.

The temperatures and differential pressures required to cause SGTR depend on the characteristics of any flaws that may exist in the tubes due to the postulated degradation

mechanisms (e.g., axial or circumferential stress corrosion cracking or damage from loose parts). NUREG/CR-6521, “Estimating Probable Flaw Distributions in PWR Steam Generator Tubes,” provides estimates for the number of flaws of each type that would be present in the SGs of lightly degraded, moderately degraded, and severely degraded SGs. For the plant analysis supporting this research effort, a moderately degraded SG as defined in NUREG/CR-6521 was used.

To evaluate the failure times of specific reactor system components, predictions of a plant’s thermal-hydraulic response and conditions that challenge the RCS components were needed. Thermal-hydraulics analyses were completed using the SCDAP/RELAP5 systems analysis code aided by computational fluid dynamic (CFD) simulations that provided local three-dimensional details of the thermal conditions at specific locations of interest. Details of the predictions and methodology used by the staff in support of the research to support closure of the SGAP are discussed in NUREG/CR-6995, “SCDAP/RELAP5 Thermal-Hydraulic Evaluations of the Potential for Containment Bypass During Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR” (to be published during 2009). The staff’s thermal-hydraulic evaluation focused on severe accident scenarios that resulted from SBO events in Westinghouse four-loop PWRs. The scenarios that challenged the SG tube’s integrity involved either a counter-current natural circulation flow pattern during conditions referred to as high-dry-low or full-loop natural circulation flows (which are possible if the water in the loop seal and lower downcomer is cleared). During the latter scenario, the mass flow and heat-transfer rates, along with the amount of mixing and entrainment between the hottest and cooler flows, are significant factors in determining the timing of the RCS failures. If loop seals are cleared and full loop natural circulation is established, the challenge to the SG tubes increases because hot steam from the reactor vessel enters the SG tubes without the benefit of counter-flow cooling. However, the most recent thermal-hydraulics analysis indicates a low likelihood of loop seal clearing. Specifically, loop-clearing behavior is predicted only during conditions where RCP seal leakage exceeds 400 gpm/pump.

Failure of SG tubes under severe accident conditions was modeled by a combination of creep rupture and limit load analyses. Tubes with flaws that are part-through-wall are assumed to fail by creep failure of the remaining ligament. The initial ligament pop-through failure is then followed by either a widening of the opening due to continued creep or by a sudden rupture if a limit load failure condition is reached. Tubes with an initial through-wall flaw are assumed to have the flaw widen either by creep or by sudden rupture if the limit load condition is reached. Ideally, the aggregate crack opening area required for a C-SGTR would be determined by a detailed severe accident analysis; however, in the absence of such a study, it was assumed that (a) flow through the cracks is choked, and (b) early containment bypass occurs if the contents of the RCS would be released through the cracks in less than 4 hours[2]. An uncertainty distribution for the required crack opening area was determined by considering the uncertainties in (1) the

[2] The four-hour time frame was chosen as a surrogate measure for an early containment bypass measure (i.e. a release occurring before the evacuation of the public is fully implemented.)

release time for containment bypass, (2) the temperature of the gas exiting the break, (3) the specific heat ratio for the gas mixture, and (4) the average molecular weight for the gas mixture. Using this analytical approach, the mean crack opening area for a containment bypass event

was calculated to be 0.081 in^2. The lower and upper 90-percent confidence limits for this value were calculated to be 0.053 in^2 and 0.124 in^2, respectively.

If a structural failure were to occur in the SG tubes during a postulated severe accident, containment bypass could occur (e.g., failure of a tube, or multiple tubes, which releases radioactivity from the RCS into the SG secondary coolant system from which it may escape to the environment through the pressure relief valves). However, if other RCS components inside containment fail before the SG tubes, containment bypass should be averted because the differential pressure driving force that would lead to such a SGTR would be reduced. Therefore, the relative timing of the potential failure of RCS components is significant because the potential risk of containment bypass would be eliminated or significantly reduced if it could be demonstrated that the primary side is either fully or significantly depressurized prior to significant heating of SG tubes. Such mitigating events may be caused by a breach in the passive RCS components due to rupture of the hot-leg piping, pressurizer surge line, SG manways, and instrument lines; or, by a failure of one or more of the active components such as the RCP seals, power-operated relief valves (PORVs), or pressurizer SRVs. Of these potential mitigating events, the behaviors of RCP seals, PORVs, and SRVs during severe accidents are the more difficult to quantifiably predict because experimental data relevant to the behavior of these components under severe accident conditions are not available. As such, RES examined PWR RCS components under severe accident conditions to better understand the likelihood of failure of these components that could prevent or mitigate a postulated containment bypass.

The final report (Reference 3) summarizes the systematic examination of PWR RCS components (other than SGTs) under severe accident conditions. The research consisted of three phases:

- Phase I reviewed methods and models for predicting failure modes and times-to-failure, identified additional information needed for the study, and scoped RCS components that might fail before SG tubes, thereby eliminating or reducing the likelihood of offsite radioactive releases by precluding SG tube failure and/or significantly reducing the RCS pressure. RES held a workshop—attended by various experts from industry, Argonne National Laboratory (ANL), EPRI, and NRC—that was used to assist in identifying focus areas related to the potential behavior of RCS relief valves and flanged connections under severe accident conditions. The results from Phase I were used to structure the approach and scope of Phases II and III.

- During Phase II, three-dimensional computer models of selected active and passive components were developed for a representative Westinghouse 4-Loop plant. A decommissioned plant (e.g., Zion Nuclear Power Station) was selected for this purpose. Detailed mechanical and structural drawings were obtained from archival records for the selected components and, based on these drawings, detailed analyses were performed. In addition, reviews were performed—using several data bases—of the operating history of relief valves, bolted and flanged connections, and spiral-wound gaskets. The component models developed in Phase II were analyzed in Phase III.

- During Phase III, several improvements were made in the thermal-hydraulics modeling using the RELAP5 code and CFD analyses, including refining the RELAP5 results to account for entrance effects and flow reversals during PORV cycling. In addition, high-temperature tensile and creep tests were conducted on stainless steel and carbon steel weldments, which extended the scope of the high-temperature materials database. These enhancements provided for a more realistic calculation of the failure sequences of the various RCS components, including determining that, in the event of a postulated severe accident, the RCS hot leg (HL) piping would most likely fail before the surge line and the RCP seals could possibly fail prior to an unflawed tube in the hottest region of the SG[3].

  If such a HL failure occurs first, it would remove or reduce the challenges to the SG tubes and preclude occurrence of a C-SGTR. The most recent predictions indicate that it requires the addition of a stress multiplier in the range of 1.5 to 2.0 (e.g., existence of flaws in the tubes) to have the hottest SG tube fail just prior to the HL for a case that assumed nominal RCP seal leakage rates of 21 gpm/pump. Because SGs in operating plants may have flawed tubes, this conclusion indicates that conditional C-SGTR probabilities must be estimated using probabilistic methods for RCS conditions and tube flaw assumptions.

To facilitate the assessment of conditional failure probabilities, SG tube and RCS component failure models were programmed into an Excel spreadsheet. Uncertainty distributions for key model inputs were then developed using an Excel add-in, Crystal Ball. The Crystal Ball software samples from the uncertainty distributions for the input parameters and runs Monte Carlo analyses to generate probability distributions for key model outputs. Model parameters for which uncertainty distributions were considered include the following:

- The length and depth of each flaw.
- The Larson-Miller parameter.
- The stress magnification factor.
- The axial location of the flaw.
- The tube inlet temperature.
- The crack area required for containment bypass.
- The hot leg and surge line failure time.

The Monte Carlo analysis was performed in three steps:

1. Calculate the time at which the steam generator tube critical crack opening area is reached.
2. Sample from the uncertainty distributions for the hot leg and surge line failure times.

[3] It should be noted that the analyses related to HL and surge line may be applicable to other PWRs provided the relevant dimensions and constraints are comparable and the analysis performed for RCS components was of sufficient quality to be compared to the analysis performed on SG tubes for a given thermal hydraulic transient and was sufficiently generic as to be applicable to other Westinghouse plants.

3. If the critical crack opening time is before the earlier of the hot leg or surge line failure times, assume that containment bypass would have occurred.

Steps 1 through 3 are repeated until sufficient samples have been run for an adequate estimate of the containment bypass probability. The bypass probability is the fraction of the Monte Carlo samples in which containment bypass is predicted to occur.

For the most challenging sequences for the SG tubes, the conditional failure probability for a SG tube was estimated to be 0.4. This probability compares well with the worst-case probabilities of 0.3 (RES), 0.2 (NRR), and 0.5 (RES bounding) given in Table 5.6 in NUREG-1570 (Reference 5) for an "average plant" as defined in the NUREG.

The potential for containment bypass under the high-dry-low conditions is effectively eliminated if the:

- RCS pressure is reduced because of operator actions or primary system leakage (eliminating the high-pressure condition);
- Feedwater flow is maintained (eliminating the dry condition); or,
- SG secondary system retains pressure (eliminating the low pressure condition).

A key insight obtained from the supporting work in References 3 and 4 is that, even if the HL failure does not occur before some degree of C-SGTR occurs, a high likelihood exists that an HL failure will closely follow a C-SGTR. This would not prevent containment bypass but would significantly reduce the flow of fission products out of the breach, thus mitigating the severity of a fission product release into the atmosphere. This benefit is not accounted for or credited in the PRA work of Reference 1 because the RES C-SGTR PRA report focuses on estimating the frequency of containment bypass due to C-SGTR rather than estimating large early-release frequency (LERF) calculations.

Conclusions

Based on the results from the example applications of the assessment method, it can be concluded that the contribution of C-SGTR events to the overall containment bypass frequency is lower than, or at the same order of magnitude as, the containment bypass fraction due to other internal events for most current PWRs. This implies that C-SGTR represents neither a negligible nor an excessive contribution to total plant risk. Therefore, C-SGTR risk should be considered and monitored in plant risk assessments in a manner commensurate with its expected importance for each plant. An additional finding is about the C-SGTR sequences that may be directly initiated by the nature of the initiating event (this mainly refers to accident sequences resulting in pressure-induced C-SGTRs). However, the contribution of such sequences to the plant risk appears to be much smaller than those sequences that are already progressing as core damage and may have the characteristic to challenge the steam generator tubes.

In light of these conclusions, and on the basis of studies accumulated over the years for the

C-SGTR, the following PRA-related technical recommendations are offered:

- Plant PRAs should address C-SGTR in their evaluation of consequences of containment bypass (LERF or level 2 analyses) on a plant-specific basis in accordance with the existing ASME PRA standard.
- Because dominant accident sequences that are potential precursors of C-SGTR are well defined and identified, they can be collected and processed for their contribution to containment bypass using the straightforward method in this report. Therefore, an in-depth and intrusive modeling of C-SGTR within the current level 1 PRA models may not be necessary.

- Care must be taken not to take excessive credit for mitigative actions, other recovery actions, and non-safety-related equipment to avoid C-SGTR, especially for external event scenarios, to obtain as realistic a risk estimate as possible.

This RES study included several limitations that should be considered:

- No explicit calculations are made for the conditional C-SGTR probability of SG tubes in Combustion Engineering plants. Because the geometry of these SGs are different from those of Westinghouse SGs (e.g., less distance between the SG inlet and the lowest point of the tubes in CE SGs, thus less opportunity for mixing of the hot water coming into the lower SG region), this failure probability may be higher for CE plants. This is acknowledged in the results but does not affect the methodology approach.

- No LERF analysis is performed. The containment bypass frequencies are estimated and presented. The containment bypass frequency provides a bounding estimate for LERF. Additionally, recent analyses supporting the State of the Art Reactor Consequence Analysis (SOARCA) project indicate that the releases from a C-SGTR are significantly lower than previously assumed due in large part to updated steam generator decontamination factors and the potential for RCS hot leg failure (Reference 8).

Although it is not expected that addressing these limitations would substantively change the methodological approach for assessing C-SGTR risk, additional studies would provide more realistic estimates of conditional RCS component and SG tube failure probabilities. Possible follow-on work includes:

- Development of updated RCS component and SG tube flaw distributions.

- Realistic three-dimensional modeling of components that better accounts for component geometry, materials, heat-transfer coefficients, and temperature distributions (e.g., ABAQUS modeling).

- Updated conditional component failure probabilities based on more recent thermal-hydraulic and materials studies.

Regulatory Implications

The current standard approach for addressing conditional SG tube failures is contained in NUREG-1570 (Reference 5). However, NUREG-1570 does not reflect the current state of knowledge in thermal-hydraulic analysis, component performance assessment, and recent industry efforts to replace SGs. SGAP Item 3.12 requires an assessment of the need for developing new guidance for the evaluation of C-SGTR risk. The results of the research studies described in this RIL could form the foundation for new guidance in this area. However, future guidance for C-SGTR risk will need to address the plant specific nature of these phenomena and comport with existing PRA standards.

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