Tools and Methods for Assessing the Risk Associated with Consequential Steam Generator Tube Rupture

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Abstract: Accidents involving SG tube ruptures can significantly contribute to plant risk, mainly because of their potential for radioactivity releases outside containment (containment bypass scenarios). The latest study of the risk associated with CSGTR has been completed and it was issued for public review[1]. The final document which includes the resolution of public comments is under publication by NRC (NUREG-2195(1)). This study involved development of new tools and guidelines to support Probabilistic Risk Assessments (PRA), and provided detailed risk insights based on the application to two US PWR designs(2). This paper provides a technical summary of the methods and approaches that directly supported the development of PRA for CSGTR study as documented in NUREG-2195. The paper provides a detailed discussion on the software called CSGTR Calculator (3). CSGTR Calculator utilized the latest available models for estimating the failure probability of SG tubes, and major RCS components (i.e. hot leg and surge line). The CSGTR failure probability and the resulting primary to secondary leak area as a function of accident time are estimated by the CSGTR calculator and are directly used in developing CSGTR Level 1 PRA. The failure probability of either a hotleg or the surge line at different times during accident is used to determine containment bypass probability for Level 2 PRA. Extended/bridge event tree, to determine the LERF (Large Early Release Frequency) resulting from CSGTR, is examined. The insights gained from the latest TH analyses using MELCOR and SCDAP/RELAP5 codes, as they pertain to developing Level 1 and 2 PRA models, are also discussed. With the publication of NUREG-2195 expected within this year, this paper can be considered as a summary guide for developing CSGTR PRA. This paper may be consulted prior to the reader refers to the NUREG document for more detailed information. This article mainly focusses on a portion of the NUREG document which deals with the PRA methods, data, and supporting tools. It does not discuss other more deterministic areas covered in the NUREG.

Keywords: PRA, Severe Accidents, LERF, Software

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) and industry have expended considerable resources over the last two decades to better understand the safety implications and risk associated with consequential steam generator tube rupture (CSGTR) events; i.e., events in which steam generator (SG) tubes leak or fail as a consequence of the high differential pressures and/or elevated temperatures during accident sequences. In an earlier study NUREG-1570(4), it was assumed that SG tubes could fail prior to the failure of other reactor coolant system components and result in containment bypass via CSGTR. SG tube integrity research has continued to be sponsored by NRC at Argonne National Laboratory and other organizations after publication NUREG-1570.

The study documented in NUREG-2195 with the expected publication within this year (2018), is different from previous studies in the following areas:

- Considered more typical replacement SG tube materials such as thermally treated Alloy 600 and 690

[1] At the time of writing of this paper, the NUREG was undergoing NRC publication process. The latest publicly available version, Draft NUREG-2195, can be found in ADAMS with Accession No. ML17082A326.

- Used recent industry data on flaw type and flaw sizes to establish a sound statistical basis[5] for generating flaws for CSGTR code

- Use of CSGTR software to predict the time and leakage areas from failed SG tubes, empirical models for failures of hotleg and surge line, and formal treatment of uncertainties.

- Use of MELCOR severe accident code for an example Combustion Engineering (CE) design[1, 6] plant, to get some insights about the releases due to CSGTR.

- Use of SCDAP/RELAP5 for an example Westinghouse (W) plant[7, 8] for detailed evaluation of thermal-hydraulic condition at and shortly after the onset of core damage.

- Use of finite element analysis for the failure of RCS HL nozzle to verify the reasonableness of the empirical models[1].

The work in NUREG-2195 includes the results of the latest research studies which covers thermal hydraulic analyses of station black out scenarios (SBOs), failure assessment for example RCS components using finite element models, time dependent failure probability and leakage areas from flawed SG tubes, and methods for including the CSGTR into an existing PRA. The PRA approach addresses the possible changes in Level 1 and 2 PRAs for evaluating the Large Early Release Frequency (LERF) due to CSGTR.

1.2 Objectives

The objective of this article is to provide a technical summary of the methods, tools, and data required to evaluate the risk associated with CSGTR. These discussions also include the description and uses of the CSGTR calculator; the software tool for calculating the probability and leak area for SG tube failures. Insights from accident analysis in support of developing both Level 1 and 2 PRA models capable of estimating the core damage frequency and the LERF contribution from CSGTR are also delineated.

2. CSGTR CALCULATOR SOFTWARE

The CSGTR Calculator utilizes the latest test results and mathematical models from Argonne national laboratory to predict the failure probability and the associated leak rate of flawed SG tubes. It also uses EPRI empirical correlations for failure of hot leg and surge line. Detailed compilation of the technical basis for CSGTR calculator routines are documented [3]. CSGTR software relies on other computer routines for pre-processing of input and post-processing of the output files to support the development of a PRA that includes risk from CSGTR. The Calculator utilizes the most recent plant data and experimental test results. It can model tubes made from thermally treated Inconel 600 and 690, as well as the mill annealed Inconel 600. The CSGTR calculator models both types of failures; creep ruptures (CR) and pressure induced ruptures (PIR). The set of equations describing the CR failures and the associated leak areas are different from those for PIR failures. The existing test data suggests that at high temperatures (temperatures greater than 600°C), the CR model is a better predictor of the tube failure time and should be used in lieu of the pressure induced models. The calculator requires an input file on existing flaws on SGs. This input file can be developed using actual flaws existing on each SGs as identified by the routine inspections, or simulated flaws based on the overall historical data on flaws found in US plants. For the latter, a series of flaws are simulated using the flaw statistics[5] for the SG tubes that are used as input to the calculator.

The CSGTR calculator currently models two types of stylized flaws: cracks and wears (volumetric flaws). Volumetric flaws are represented by the SG tube degradations, such as wall thinning, denting, pitting, etc. All flaws were considered standardized. Volumetric flaws were all assumed to be smooth

and characterized by a length or depth. The effect of the width for wear flaws is not considered; i.e. the width was assumed to be large. Cracks were stylized as single circumferential or axial cracks, each with a length and a depth. Multiple shallow cracks are not explicitly modeled although they can be simulated by an approximate representation via the two types of stylized cracks. The uncertainty associated with assuming the idealized flaw shapes on failure probabilities were considered by assigning uncertainty factors to the measured flaw size data.

CSGTR software is written in Java language. It is probabilistic in nature, and trials are generated using Monte Carlo sampling. A trial constitutes one set of samples taken from all uncertain parameters. An uncertainty distribution is assigned to all the variables used within the code. Uncertainties associated with the thermal-hydraulic (TH) results can be handled external to the code, via multiple runs each with different TH input file.

For each trial, the inner core of the software performs the analysis, progressing through all time steps. The steps involved in performing these calculations are described below:

Step 1: At the beginning of each time step, all the information from previous time step is carried forward. The information include the condition of each of the flawed tubes (intact, leaked, or ruptures), the associated leak area, and the amount of cumulative creep integral. At zero time step, none of the flawed tube is considered failed and the creep integral is set to zero.

Step 2: From the tube wall temperature, decide which model to use; CR or PIR. The code switches from PIR mechanism to a CR mode based on the tube threshold temperature (i.e., temperature above 600°C).

Step 3: When PIR model is used, the code determines if the tube is remained intact, failed and leaking, or failed and ruptured. When tube is ruptured, the leak area is calculated based on rupture model (also known as the fish mouth opening). A different model is used for leaking tubes (also known as wall through failure).

Step 4: When the CR mechanism is applicable the program increases the leak area due to creep and evaluates the failure models for hot leg and surge line to determine the size of the leakage through SG tubes at the time of RCS failure.

Step 5: The code then maintains the information for that time step to be used as status for the next time step. A new cycle of calculations the will start (see Step 1) until all the time steps are accounted for.

Step 6: Considering steps 4 and 5, the software will track the equivalent SG leak size at each time step until catastrophic failure of either hot leg or surge line. In this manner the code track how the SG leak grows as a function of accident time.

In the outer loop, the program performs a large number of trials (random samples) to establish the statistics and the distribution for output parameters. These statistics are used to estimate the tube failure probabilities and the leak probability distribution for each tube and the cumulative leak probability distribution function for all flawed tubes at each time step.

The uncertainty distributions were defined based on the information in open literature. They are generally expressed in three forms: multiplicative, additive, or special form. The samples for multiplicative uncertainties (T) are generated by the predicted value multiplied by (1+ error term). The additive uncertainties are generated by the predicted value added by an error term. Some uncertainties were expressed in a special format where a random variable was used within the empirical equation for model prediction (CR models for Inconel 600 and 690). The model that best described the uncertainties for various ranges of parameters was selected. The uncertainty parameters are maintained

in a protected input file (library file); however it can be updated if needed. Table-1 shows the sources of uncertainties considered in CSGTR software.

Table-1 Sources of uncertainties considered in CSGTR software

Sources of uncertainties in CSGTR
All dimensions and measurements: SG tube inner radius, tube thickness, hotleg inner radius hot leg thickness, surge line inner radius, surge line thickness
Material properties: the Poisson ratio, the yield stress, the ultimate stress, the flow stress coefficient, the Young's Modulus of Elasticity as a function of temperature for each type of material (Inconel 600, 690, SS304, carbon steel, etc.)
Flaw characteristic sizes: Flaw length and depth
Empirical model uncertainties: a) axial burst and leak pressure b) circumferential burst and leak pressure c) axial burst and leak areas d) circumferential burst and leak area e) CR time to failure f) CR leak area

The CSGTR software generates three different files, DEBUG file, Intermediate file, and cumulative file. Debug file is a large file which contains time step/trial based information. It was intended to be used by the program developers or advanced users. It is often suppressed and not generated.

The intermediate file is a statistical summary file. It contains the following information at each time step for each flaw.

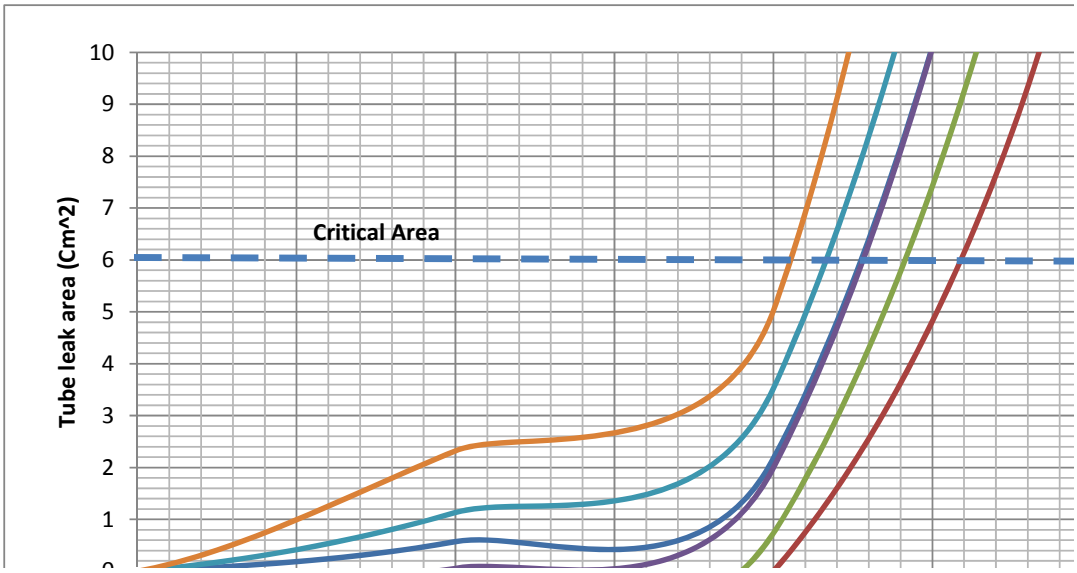
1. Leak Probability due to pressure induced
2. Burst Probability due to pressure induced
3. Probability of failure due to creep rupture
4. Average Burst Area due to pressure induced, and its 5 and 95 percentiles
5. Average Leak Area due to pressure induced, and its 5 and 95 percentiles
6. Average Leak Area due to Creep Rupture and its 5 and 95 percentiles
7. Failure probability of hot leg as function of time
8. Failure probability of surge line as a function of time

The Cumulative Leak Area file is a text file generated from the Intermediate file. The cumulative leak area is calculated by summing the leak area across all the flaws (i.e., and all SGs).

Figure-1 shows an example of the results generated by the CSGTR software. In this graph the cumulative SG leak area through the SG tubes is shown as a function of accident time. Figure-1 also illustrates the uncertainty predicted for the results by showing the noted percentiles of the uncertainty distribution. For this case, the mean leakage area (A_m) is very close to the 50 percentile of the distribution (A0.50) throughout the simulated accident. These uncertainties do not include the uncertainties associated with the accident simulation (i.e. TH uncertainties).

The failure probabilities of hot leg and surge line failures are reported separately. This information is used to generate the RCS survival probability. This is done as a part of post processing routines which also estimate the containment bypass probability. Figure-2 shows an example of type of results generated by post processing. This figure shows the probability that RCS survive[2] as a function of time and the probability that SG leak rate is less than some value (in this case the graph shows leak area of 3 cm^2, and 6 cm^2, respectively). This latter probability is estimated by using the full uncertainty distribution of SG leak rate sizes as a function of accident time. The containment bypass probability then can be estimated by the probability that CSGTR leak rate exceeds a critical leak[3] area multiplied by the probability that RCS survives as a function of accident time.

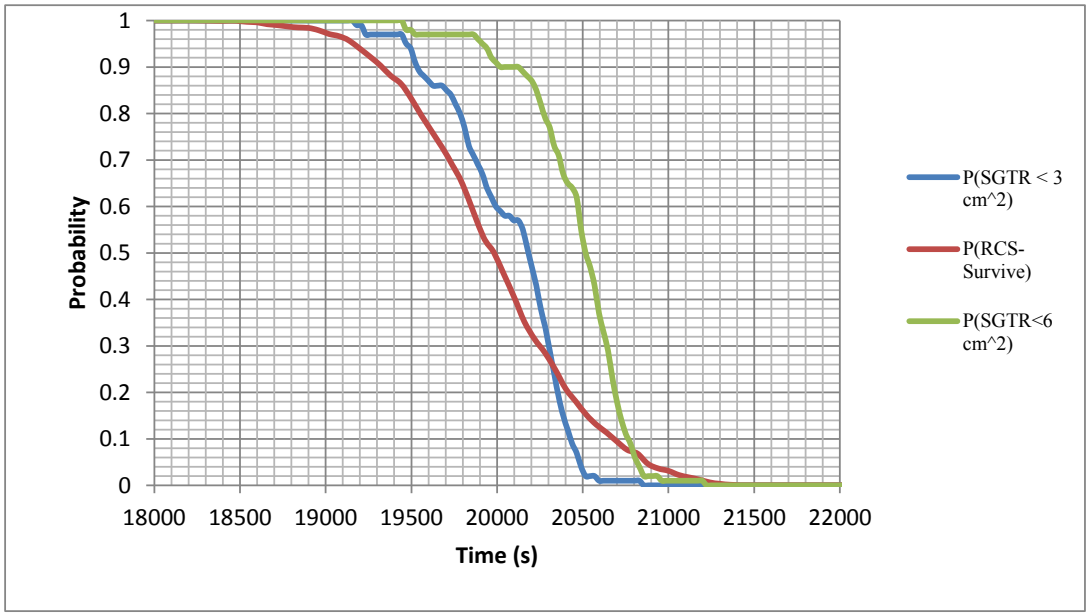
Figure-1 Percentiles of the SG leak area distribution as a function of accident time



[2] The probability that RCS survive is defined by the probabilities that hot leg and surge line do not fail. This is calculated by one minus the sum of the failure probability of any of the hot legs (e.g., four times the failure probability of one hot leg in a four loop plant) and the failure probability of surge line.

[3] The critical leak area is chosen by analyst and it is an integrated leak area that defines CSGTR, when reached before gross RCS leakage (e.g. hot leg failure) occurs.

Figure-2 Cumulative SG leak area and the probability that RCS remains intact as a function of accident time



3. CSGTR-PRA APPROACH

CSGTRs event could occur before or after the onset of core damage. CSGTRs before onset of core damage are usually due to PIR and they will be modelled within the level 1 PRA. CSGTRs occur post core damage (i.e., generally due to CR) will be modelled as a part of the Level 2 PRA. When Level 2 PRA is not available simple models are used for evaluating changes in LERF.

Identification of candidate sequences for CSGTR challenge before or after onset of core damage is relatively straightforward, given the existence of a Level 1 PRA model. Furthermore, a bounding evaluation of core damage and LERF due to CSGTR can be performed with the existing level 1 PRA model (with minor changes when required).

Possible changes to level 1 and level 2 PRA models for more refined analysis are discussed next. . The following discussions therefore should not be regarded as a recommendation for extensive additions to an existing level 1 PRA model.

3.1 Changes in PRA for Incorporating the PIR failures of SG tubes

There are initiating events and failure of safety systems during an accident sequence that could create a delta pressure across the SG tube walls and, therefore, potentially challenge the integrity of the tubes because of PIR mechanisms. The first step of the analysis is to select these scenarios from level 1 PRA which can have potential for CSGTR due to PIR. A general understanding of the TH behaviour of the primary and secondary loops to determine the possible ranges of delta pressure across the tubes is essential for this task. Table-2 provides a generic list of these sequences. The sequences identified in Table-2 could be part of the internal or the external event PRA. Some scenarios such as ATWS are more dominant for the internal event PRA, some like spurious opening of SG PORVs could become more dominant during fire initiators, and some like secondary side break (SSB) may become more dominant during seismic events. The occurrence of SG tube failure due to PIR in these sequences can impact the accident progression path. Depending on the size of leak rate, there could be additional need for injection. These events behave similar to SGTR and may require additional actions for

isolation of the faulted SG and refilling the RWST tank if needed. The timing for the accident progression could be somewhat different than SGTR modelled as an initiator in PRAs. Performing additional TH calculation may become necessary to estimate the revised timing for operator actions based on the size of SG leak which depends on the number and the sizes of the flaws. To simplify the PRA evaluation, one critical leak size equivalent to twice the cross sectional area of one tube (i.e. guillotine break of one SG tube) is considered. A set of example scenarios are analysed for CSGTR due to PIR and are detailed in Section 7 of NUREG 2195[1].

Standard level 2 or LERF methods used for SGTR can be directly used for CSGTR events due to PIR. For example all sequences resulting in early core damage before effective evacuation with dry condition in SG secondary side (non-flooded condition) could be considered as LERF.

3.2 Changes in PRA for Incorporating the CR failures of SG tubes

The best way to identify the core damage scenarios that can potentially challenge SG tubes during severe accidents is to use the binning information generated for defining plant damage states (PDS) from Level 2 PRAs. Those PDS that are binned into a class with a high primary pressure, and with at least one or more dry SGs are candidates for severe accidents with a potential for CSGTR. However, for all other plants without a Level 2 PRA models, a simple level 2 bridge event tree is required. The L2 bridge tree is similar to, and may be combined with, the Level 1 extended event tree. Generic insights gained from TH analysis in NUREG-2195 and other studies can be used to develop the Level 2 bridge tree for CSGTR.

Generally, the goal of extended Level 1 event tree is to specify information needed for PDS binning as starting point for level 2 analyses. Extended Level 1 event tree determines such information as the status of containment systems, primary and secondary loop conditions, starting time of core damage, and all other possible failures of barriers that can result in direct release path to environment (such as Interfacing System LOCAs; ISLOCAs and SGTR). The goal of Level 2 bridge tree is to specify the status of all systems, including containment system, primary and secondary conditions, possible occurrence of CSGTR, and all other failure of barriers that can result in a direct path to release at or shortly after the onset of core damage. Therefore, there could be quite an overlap between L1 extended event tree and the L2 bridge event tree. The major differences are the failure and the leak area of SG tubes, the failure probabilities of hot leg and surge line, and the status of other relevant conditions such as SG relief valves, PORVs, primary and secondary pressure, and loop seal clearing. The L2 bridge event tree is limited to the onset of the core damage (i.e., not for other phases of core melt progression such as in-vessel and ex-vessel melt progression [9]). It is used as the basis for binning of the plant damage states when dealing with CSGTR due to CR. The L1 extended event tree and L2 bridge tree share common structure and could be combined into one event tree.

The focus of discussion on the L2 bridge event tree in this article is for the CSGTR due to CR. The other parts that consider being an overlap with L1 extended event tree is not discussed.

3.2.1 Bridge L2 Event Tree for CSGTR due to CR

For CSGTR occurs due to CR the following conditions have to be satisfied:

1. Onset of core damage has occurred and it is detected via either core exit temperature or start of additional heat generated by Zircalloy oxidation
2. Loss of feed to one or more SGs has occurred (one or more dry SGs)
3. Primary pressure is high; high pressure is defined as a pressure above ~1200 psi which is sufficient to open the SG secondary relief valves if CSGTR occurs
4. Secondary pressure low prior to occurrence of CSGTR: it is always assumed to be the case for dried SGs due to nominal leakages through secondary loop.

All Level-1 core damage sequences will be linked to the CSGTR bridge tree. Loss of feed to one or more SGs (question 2) can be modeled via a fault tree which models that at least the feed to one SG is lost.

Table-2 Selected DBA Sequences Causing Challenges to SG Tubes

Delta P Across the Tubes	Conditions Causing Delta P Across the Tubes	Accident Sequence	SG Secondary-Side Condition [Pressure, Water Inventory]
~ 6.9 mega pascals (MPa) (1,000 pounds per square inch (in.) (psi))	Normal power operation	SGTR event	Not known, will be determined by resulting CD sequences
~ 10.34 MPa (1,500 psi)	FB sequences with medium-head [a] emergency core cooling system (ECCS) pumps	All sequences involving loss of secondary heat removal but success of FB	Low pressure and dry SGs before rupture Low pressure and dry SG condition is expected after CD
~ 10.34 to 11.75 MPa (1,500–1,700 psi)	1. Unisolable main steamline breaks (MSLB) 2. Spurious opening of SG relief valves or turbine bypass valves with failure to isolate	All sequences are expected to result in loss of secondary cooling followed by FB cooling.	Low pressure but not dry SGs before rupture Low pressure and possibly dry SG condition after CD
~15.5 MPa (2,250 psi)	FB with high-pressure ECCS pumps	All sequences involving loss of secondary heat removal but success of FB with stuck-open secondary relief valves	Low pressure and dry SGs before rupture Low pressure and dry SG condition after CD
22.06 MPa (~3,200 psi)	ATWS sequences when secondary cooling is lost and pressure peak is limited to <22.06 MPa (<3,200 psi)	ATWS sequences with a favorable moderator temperature coefficient can result in a pressure peak as high as 22.06 MPa (3,200 psi) in the primary.	High pressure and dry; however, failure of SG tube will induce CD. All such CD sequences during ATWS are treated as LERF.

[a] The ECCS pumps used in U.S. PWRs can have a shutoff head as low as 8.27 MPa (1,200 psi) and as high as 18.27 MPa (2,650 psi).

[b]UET (unfavorable exposure time) is defined as the time during the cycle when the reactivity feedback is not sufficient to prevent RCS pressure from exceeding 22.06 MPa (3,200 psi). Many factors, such as initial power level, time in cycle when transient occurs, reactivity feedback as a function of the cycle life, the number of available primary relief/safety valves, the failure or success of control rod insertion, and auxiliary feedwater (AFW) flow rates, affect UET. The noted pressure below 22.06 MPa (3,200 psi) is used as the bounding primary pressure value for cases when the moderator temperature coefficient is <u>favorable</u>.

This is shown usually with a different heading; for example AFW-T on the bridge tree, to differentiate it from the AFW fault tree for L1 PRA which generally models loss of feed to all SGs. PRAs generally include recovery actions for secondary side cooling to at least one SG. The recovery action for extended L2 event tree is valid only if feed is recovered and established to all SGs. Recovery actions may be omitted for most scenarios except SBO scenarios.

High primary pressure (question 3) is generally modeled by a fault tree and with the aid of a set of linked event tree rules. It is clear that for events such as Large/Medium Loss of Coolant Accidents (L/M LOCAs), the primary pressure would be maintained low. These initiators including large ISLOCAs can also be excluded when specifying high primary pressure. There are also several scenarios in L1 PRA that primary pressure will be low before onset of core damage but has the

potential for becoming a high primary pressure scenario at or shortly after the onset of core damage. These include sequences when the primary is depressurized initially by opening one PORV, or by using the secondary system for rapid cooldown and depressurization (with available SGs, not necessary all). TH analysis for both RELAP/SCDAP for example Westinghouse plant and MELCOR for the example CE plant showed that; to maintain a depressurized primary condition post on set of core damage, all PORVs (two or more) are required to be open. This indicates that the success criteria for primary depressurization for the L2 bridge event tree are more stringent than the primary depressurization success criteria during L1 event tree. A fault tree could be developed to represent a specific plant condition by accounting for its number of PORVs and L1/L2 success criteria. The depressurized state of the primary system can be modeled with a combination of linked event tree rules and a fault tree. For sequences with SLOCA, the success criteria in some cases can be relaxed; requiring a lesser number of PORVs to open.

The L2 bridge tree would also contain nodes that help defining the magnitude of radioactive releases. This is especially important in determining the LERF outcome. This is discussed next.

3.2.2 LERF Model for CSGTR due to CR

Using the CSGTR bridge tree, we can determine contribution of all sequences that can potentially contribute to LERF due to CSGTR. These sequences can be binned for the purpose of selecting representative TH evaluation. Table 3 shows examples for selected TH analyses. The results of TH analysis are then used as input to CSGTR software to estimate the probability of containment bypass and determination of the probability of LERF outcome.

Table3 Selected CR Sequences (sequence bins) Causing CR Challenges to SG Tubes

Core Damage Sequences	Notes
Base Case-1: SBO with failure of TDAFW at time zero, small RCP leakage[a,b] (e.g., 0.001325 m^3/s (21 gpm) for some W pumps), and equivalent 0.0127 meters (0.5 inches) of leakage (relief path) from the SG secondary to the environment	Base case probability of CSGTR before HL failure
Alternate case-1: Base Case-1 and 1 PORV or an SRV sticks open	Lower probability of CSGTR before HL failure due to lower primary pressure
Alternate case-2: Base Case-1 except RCP seal leakage greater than 0.01136 m^3/s (180 gpm) per pump	Possibly higher probability of CSGTR due to possible clearing of the loop seals
Alternative case-3: Base Case-1 except larger leak area through the secondary of SGs (e.g., as a result of a stuck-open SG PORV)	Higher probability of CSGTR assumed since, after tube failure, the larger area through SG secondary would depressurize the primary, and therefore, reduce the likelihood of HL failure
Base case-2: SBO with failure to load shed to extend battery life, rendering the failure of TDAFW to continue to run	Similar probability of Base Case-1
Base Case-3: SBO with successful load shed to extend battery life. Failure of TDAFW to continue to run after battery depletion	Similar CSGTR probability of Base Case-1
Base Case-4: with total failure of secondary cooling at time zero and failure to do FB operation	Similar probability to Base Case-1
Alternate Case-4A: SLOCA and transient sequences with partial loss of secondary cooling such that at least one SG is dried up.	Possibly lower probability of CSGTR due to partial cooling and later time associated with release
Base Case-5: SLOCA and transient sequences [c] with delayed failure of secondary cooling and FB operation (e.g., loss of service water, loss of chilled water due to external hazards]	Varying probability of CSGTR, depending on plant-specific features and the details of the sequences. These sequences could also cause RCP seal failures, with varying degrees of leakages.

Core Damage Sequences	Notes
[a] RCP leakage for base case-1 SBO analyses was set at 0.001325 m^3/s (21 gallons per minute) per pump. This is typical of a W plant without SHIELD®[2]. CE plants with Flowserve or similar seals and W plants with SHIELD® mechanical seals or Flowserve pumps or CE plants have very low leakages following SBOs. [b] Seal LOCAs for CE plants could occur as a result of loss of cooling and failure of the operator to trip the pumps. Seal leakages of 1,703 liters per minute (Lpm) (450 gpm) per pump could result. [c] These sequences consider that AFW flow is isolated to at least one SG, intentionally by the operator, since the associated secondary relief valve has stuck open.	

To address the LERF issue the following three questions shall be resolved (i.e. three additional headings for L2 bridge event tree).

1. What is the size distribution of leakage area (Probability vs. leak area) through ruptured tubes?

2. What are the size of leakage and the path to environment (release path)?

3. What is the time of release in comparison to effective evacuation?

CSGTR software estimates the distribution of leakage area through ruptured tubes when a specific TH input is selected. The code output can be further used to estimate the probability that cumulative leakage through a SG tube larger than a certain threshold size occur prior to hot leg or surge line failures. There are generally three sizes of leakages (question 1) that a PRA may be concerned about. These are:

- Small leakages through ruptured SG tubes below the nominal leakages through secondary piping (e.g. equivalent 0.5 inches in diameter): Such small leakage is not expected to pressurize the secondary side of SG therefore cannot demand cycling of the SG relief valves. This size of leakages are expected to result in a very small releases, since it has to go through the torturous path of secondary side and possibly scrubbed with the water in condenser.

- Medium leakages through ruptured SG tubes are those that are above the nominal leakages and capable of demanding the SG PORVs to cycle (slightly less than or about an equivalent area of guillotine break of one tube). Limited MELCOR analysis has shown that these leakages may not constitute large releases unless one or more secondary relief valves fail to reclose after cycling. The probability of failure to reclose for secondary relief valves when exposed to hot gases post core damage is comparatively higher than the nominal failure probability. Some recent studies [(10)] provide estimates for the failure probability per cycle of valve stroke. Information in reference-10 can be consulted for estimating the possibility that the one or more SG PORVs stick open early during accident.

- Large leakages through ruptured SG tubes are those that are larger than an equivalent area of a guillotine break of one tube. These sizes of leakage are expected to continuously and repeatedly demand the cycling of SG relief valves. The failure of relief valves therefore is not necessary for the release to be considered large. The exact size of the leakage area would be plant specific also depends on the SG specification. Such large leakages could occur due to failure of multiple SG tubes or possible clearing of the loop seals. Large leakages due to multiple large tube flaws are straightforwardly estimated by CSGTR software. Insights from TH analysis are used for evaluating the probability that loop seals may clear. The clearing of the loop seal eliminates the counter-current flow pattern and expose all SG tubes to hot gases with no cooling effect and mixing resulting from the counter current flow. This is expected to result in several tube failures. It is therefore considered to yield a large leakage through multiple ruptured SG tubes. Although all conditions associated with loop seal clearing are not yet well understood, the recent TH analysis indicates that the loop seal can be cleared if there

are large leakages through RCP seals or other downstream primary piping. The ranges of RCP seal leakages (per loop) that can cause loop seal to be cleared are estimated to be between 300 to 450 gpm. The probability of loop seal is cleared is estimated to be close to one at 450 gpm.

The release path considered for LERF is through the SG relief valves (i.e. stuck open SG PORVs and SRVs). No scrubbing is available since the secondary side of the SGs is assumed to be dry. Possible credits for SAMG activities for flooding the secondary side of SGs could eliminate the possibility of LERF. For a simplified LERF evaluation, no credit is taken for SAMG activities. Furthermore, the use of alternate systems (such as those offered by FLEX/B5b equipment) is not yet included in the analysis. All in-vessel releases are assumed to be transported to the environment via stuck open secondary relief valves. No credit is given to in-containment releases due to stuck open primary PORVs or other small primary leakages. The "release duration" is then defined by the time from the occurrence of CSGTR to completion of in-vessel releases (vessel rupture). The "evacuation duration" is defined by the time from the notification of general emergency to effective completion of evacuation. Notification of General Emergency can occur when there are indications that core damage is imminent. For example, notification of general emergency will occur when CSF (critical safety function) status tree indicates "Heat Sink Red" (applicable to these scenarios). Emergency Action Levels for PWRs should be consulted for additional information. The late release (question 3) is therefore defined when the start point of the release duration is significantly after the end point of effective evacuation duration. NUREG-2195[1] uses a value of less than ~10 hours after notification of general emergency for defining the early releases based on most recent MELCORE analysis of in-vessel phase.

4. CONCLUSION

The latest study of the risk associated with CSGTR has been completed and it is under publication by USNRC (NUREG-2195). A summary guidance for how to develop a PRA that includes the risk from CSGTR was discussed in this paper. The information presented in this paper is consistent with the insights gained from the research performed during the development of the NUREG. This paper also discussed CSGTR software which is a central tool for developing CSGTR PRA for both PIR and CR failure mechanisms. The article discusses how a L1&L2 PRA capable of estimating the LERF contribution from CSGTR events can be developed. As a part of this development, the insights gained from the latest TH analyses using MELCOR and SCDAP/RELAP5 codes for developing the L1 extended and L2 bridge tree for CSGTR were relied on. A simplified LERF model was suggested based on the information extracted from NUREG-2195. The information contained in this paper would be useful for PRA practitioners and developer prior to referring to voluminous NUREG document.

5. REFERENCES

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