An Introduction to Improved Uncertainty Analysis for the Probability of the Severe Accident Induced Steam Generator Tube Rupture

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1. Introduction

In steam generator tube rupture (SGTR) accidents, the unmitigated radioactive materials could be released to the environment via not isolated route, even though the containment structure is intact. These accident scenarios are categorized as “a containment bypass” among the containment failure mode categories in a Level 2 Probabilistic Safety Assessment (PSA), also a SGTR frequency significantly attributes to Large Early Release Frequency (LERF) which is a safety goal in the domestic nuclear regulation.

The SGTR could occur not just on the onset of an accident as a initiating event, but also be evolved by the thermally creep rupture on the hot steam generator tube or the high pressure difference between the primary side and the secondary side during severe accidents. It is called severe accident induced SGTR (ISGTR), and it was firstly reported in the NUREG-1150 [1]. The ISGTR could occur under the high pressure difference or the thermally high load condition on the SG tubes. Those are called the pressure induced SGTR (PI-SGTR) and the temperature induced SGTR (TI-SGTR), respectively. The ISGTR is also dealt with a bypass scenario in a Level 2 PSA, therefore, it is important to evaluate the probability of ISGTR. The NRC staff estimated the probability of an ISGTR in the Accident Progression Event Tree (APET) for the Surry nuclear power plant in the NUREG-1570 [2]. Recently, the Korean domestic nuclear regulatory agency as well as the NRC requires to consider the methodology in the NUREG-1570 on the analysis of ISGTR, moreover plant specific analysis in order to meet ASME/ANS PRA Standard [3].

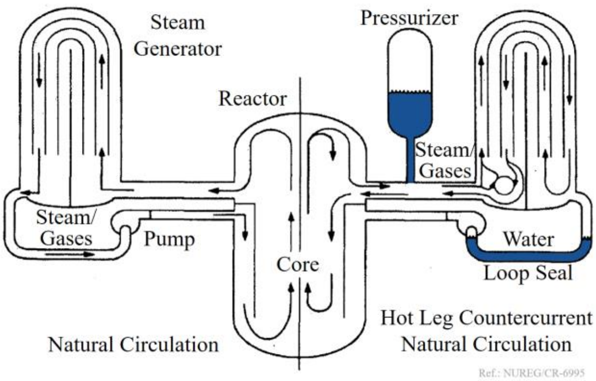

Fig. 1. Natural circulation phenomena under severe accidents in U-tube steam generator [4]

The evaluation of the probability of ISGTR in the NUREG-1570 considered Larson-Miller creep rupture model with the material properties for the Alloy 600 as the steam generator tubes, the RES (US Nuclear Regulatory Research) flaw distribution with the Monte-Carlo simulation, and the scenario dependent pressure/ temperature histories. In this paper, an improved uncertainty analysis methodology to estimate a plant specific ISGTR probability involving an uncertainty of pressure/temperature histories for accident scenarios is investigated based on the NUREG-1570.

2. The probability of ISGTR in NUREG-1570

In order to estimate a probability of an ISGTR, several thermal-hydraulic conditions and plant specific design data are required such as the pressure and the temperature of the primary side and secondary side, the flaw distribution and the material properties.

2.1. Thermal-hydraulic conditions prediction

Realistic pressure and temperature are important parameters to evaluate ISGTR, hence the various reactor coolant system (RCS) status and the SG status were considered such as the relief valve failure and the RCP seal leak. Especially the natural circulation in the RCS was focused on as a major severe accident phenomena to affect the thermal-hydraulic conditions.

In the initial status of accident a part of the core decay heat would be transported to the steam generator via hot leg pipe by full-loop natural circulation of the liquid coolant. In the status of severe accident, the hot leg and SG tubes would be voided, hence the countercurrent natural circulation of the hot gases begins. In the U-tube steam generator, the hot gases goes to the outlet plenum of the steam generator via the upper section of a hot leg and the cold gases or the liquid come back to the reactor via the lower section of the hot leg as shown in Fig. 1. The several U-tubes are carrying the hot gases, the others are carrying the cold gases. The hot gases and the cold gases are mixed in the inlet plenum of the steam generator, hence it leads to decrease the temperature of the hot gases before its entering the U-tube. The hot leg countercurrent natural circulation could occur when the intermediate loop is filled with the liquid coolant (Loop seal). If the loop seal is lost (Loop seal clearing), the hot gases is able to travel the entire RCS loop, then the superheated more hot gases enter the U-tube. The RCP seal leak is the one of the situations to lead the loss of loop seal, thereby the RCP seal leak is applicable parameter to affect the natural circulation phenomenon.

In the NUREG-1570 the thermal hydraulic analyses carried out using SCDAP/RELAP5 computer code. The several uncertainty parameters are chosen such as the mixing fraction, the recirculation ratio and the fraction of hot tubes (tube bundle split). The mixing fraction reflects the fraction of the hot gases that mixes with the cold gases in the inlet plenum of the SG. The recirculation ratio reflects the portion of the mass of cold gases mixed with hot gases raised again to the U-tube with the hot stream. The faction of hot tubes reflects the portion of the U-tubes contained the hot gases against the whole U-tubes.

2.2. Calculation of ISGTR probability and assumptions

The probability of ISGTR for the APET is described in the Table I and the Table II. Those analyses considered the Larson-Miller Parameter (LMP) for the creep model, the RES flaw distribution with the Monte-Carlo simulation, and the several relevant accident scenarios analyzed by SCDAP/RELAP5.

Table I: The Probability of TI-SGTR in APET [2]

Primary status	Secondary status (No. of SGs depressurized)	Failure probability
Intact	None	0.0173
	1 SG	0.0791
	2 SG	0.0970
	3 SG	0.1150
Stuck open PORV - Late	None	Not calculated
	1 SG	0.0184
	2 SG	0.0365
	3 SG	0.0542
Stuck open PORV - Early	None	Not calculated
	1 SG	0.0184
	2 SG	0.0365
	3 SG	0.0542
RCP seal LOCA	None	~ 0.137
	1 SG	0.392 ~ 0.401
	2 SG	0.582 ~ 0.585
	3 SG	1.0

Table II: The Probability of PI-SGTR in APET [2]

Primary status	Secondary status (No. of SGs depressurized)	Failure probability
Intact	1 SG	0.0549
	2 SG	0.107
	3 SG	0.156

The thermal-hydraulic conditions are determined by several representative accident sequences based on the several simplified assumptions. Several uncertain parameters such as the mixing fraction, the recirculation ratio, and the fraction of hot tubes set a representative value as a base case. The probability for that the number of SG depressurized exceeds one SG cases, is calculated arithmetically on the basis of the one SG depressurized case as a representative scenario. In addition, the probabilities of ISGTR in the NUREG-1570 do not contain the failure of hot leg or surge line. The NRC staff determined the probability that a flawed SG tube would fail before others component such as the hot leg or the surge line rupture [2].

3. Introduction to the methodology on the improved uncertainty analysis of ISGTR

In this paper, an improved uncertainty analysis methodology for ISGTR has been introduced based on the plant specific design and the thermal-hydraulic conditions. This methodology includes the uncertainty for a priority of occurrence on the failure of a SG tube, a hot leg, a surge line, or a reactor vessel. The calculation of the pressure/temperature history would repeat many times against various input set of the sampled phenomenological parameters. As a result, various pressure/temperature histories would affect to the timing of creep failure for each RCS component.

3.1. Uncertainty analysis for the thermal hydraulic histories

NUREG-1570 reported the major phenomenological uncertain parameters for the ISGTR such as the mixing fraction, the recirculation ratio and the fraction of hot tube as well as the RCP seal leakage rate which affects to the loop seal clearing. The MAAP code (Modular Accident Analysis Program) is applicable on such a parametric study. For example, the parameter FAOUT reflects the fraction of SG tubes carrying out flow in the hot leg natural circulation model [5]. Also MAAP5 has the hot leg natural circulation model benchmarked against Westinghouse $1/7^{th}$ scale experiment. It can mechanistically calculate the mixing fraction and the recirculation ratio in the SG inlet plenum based on a one dimensional Gaussian buoyant plume of hot gases entraining ambient fluid [5,6].

The phenomenological parameters would be sampled by Latin Hypercube Sampling (LHS) method [7]. It is an effective technique that uses less simulations than the Monte-Carlo method to realize the shape of an inherent probability distribution function (PDF) for a parameter. The range of each parameter is divided into equally probable intervals. The sample parameters are taken once from each internal. This methodology allows to avoid same degree sampling range.

3.2. Calculation of pipe stress with the flawed tubes

The hoop stress, σ, of a non-flawed pipe such as a hot leg or a surge line is calculated by its pressure difference and the geometrical data as follows:

$$\sigma = \frac{\Delta P \times R_c}{t_c} \tag{1}$$

The pressure difference, an inner radius and wall thickness of the pipe are denoted ΔP, R_c and t_c, respectively. The hoop stress of a flawed SG tube is calculated as follows:

$$\sigma = \frac{M_P \times \Delta P \times R_c}{t_c} \tag{2}$$

where, M_P is a magnification factor for part-through-wall cracks for the flawed pipe. The magnification factor is applied to an improved correlation as reported in NUREG-1570. It is related to the flaw data such as a depth and a size of a flaw.

An appropriate SG flaw distribution is required on the uncertainty analysis. The flaw distribution could be produced, if the plant specific data for the SG inspection experience is available. Otherwise, an appropriate generic data would be used. The flaw data would be taken by the Monte-Carlo simulation for each thermal-hydraulic sample set calculated by LHS based on an accident scenario.

3.3. Larson-Miller parameter and rupture data for each component

The creep data for the materials of each component are calculated by LMP correlation. The LMP is widely applied to predict creep rupture of a metallic material and its rupture time for under certain stress condition. The LMP correlation as follows:

$$LMP = T[c + d \cdot ln(t_R)] \tag{3}$$

$$ln(\sigma) = m \cdot LMP + b \tag{4}$$

where, T is a temperature of materials, t_R is a creep rupture time for a material, and the others, b, c, d and m, are LMP coefficients. These coefficients would be taken by plant specific material data for the RCS components including SG tubes.

The LMP creep data, the magnification factor of the SG tube sampled by the Monte-Carlo simulation, and the thermal-hydraulic conditions with the stress on the components for each time step calculated by MAAP code applying uncertainty parameters using Latin Hypercube Sampling are utilized as the inputs for the ISGTR calculation. The creep fraction of each component would be calculated as Eq. (5), where t_f is a life time of materials. The t_f is divided to each time step during the calculation. When the summation of creep fraction for each time step goes over unity, then the creep rupture would be occurred its component. For each simulation, if the creep rupture occurs on the SG tube at priority, the ISGTR occurs. On the other hands the creep rupture occurs on the others components at priority, the ISGTR is prevented.

$$\int_0^{t_f} \frac{dt}{t_R(T,\sigma)} = 1 \tag{5}$$

4. Conclusion

The improved uncertainty analysis methodology for the probability of the severe accident induced steam generator tube rupture was introduced in this paper. The uncertainty of the accident scenarios is additionally considered comparing with NUREG-1570. This methodology is able to calculate not only a probability of an ISGTR but also that of hot leg or surge line creep rupture. The meaning of this probability is which RCS components would fail in priority by the creep rupture. This analysis method can provide more detail results to estimate an ISGTR probability following NUREG-1570 methodology and it is expected to meet ASME/ANS PRA Standard Capability Category II.

REFERENCES

[1] NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, U.S. NRC, 1990.
[2] NUREG-1570, Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture, U.S. NRC, 1998.
[3] ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, The American Society of Mechanical Engineers, 2009.
[4] NUREG/CR-6995, C. D. Fletcher, R. M. Beaton, V. V. Palazov, D. L. Caraher and R. W. Shumway, SCDAP/RELAP5 Thermal-Hydraulic Evaluations of the Potential for Containment Bypass During Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR, U.S. NRC, 2010.
[5] T. Kindred, Modular Accident Analysis Program 5 (MAAP5) Applications Guidance: Desktop Reference for Using MAAP5 Software – Phase 3 Report, EPRI, Palo Alto, CA: 2017, 3002010658, 2017.
[6] W. A. Stewart, A. T. Pieczynski and V. Srinivas, Natural Circulation Experiments for PWR High-Pressure Accidents, EPRI Report TR-102815, 1993.
[7] M. D. McKay, R. J. Beckman and W. J. Conover, A Comparison of Three Methods for Selecting Values of Input Variables in the Analysis of Output from a Computer Code, Technometrics, Vol. 21, No. 2, pp. 239-245, 1979.