UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

May 19, 2017

Mr. Victor McCree
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: CONSEQUENTIAL STEAM GENERATOR TUBE RUPTURE

Dear Mr. McCree:

During the 643rd meeting of the Advisory Committee on Reactor Safeguards, May 4-5, 2017, we completed our review of draft NUREG-2195, "Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes." Previously, we discussed this topic during our 604th meeting, May 9-11, 2013. Our Subcommittee on Materials, Metallurgy, and Reactor Fuels also reviewed this matter on April 6, 2011, April 7, 2015, and December 15, 2016. During these meetings, we benefited from discussions with representatives of the NRC staff and their contractors. We also benefited from the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

1. NUREG-2195 should be published. The methods documented in this report have advanced the state-of-the-art for evaluating phenomena that contribute to consequential failures of steam generator tubes, which can be significant contributors to risk of offsite radionuclide releases.

2. These methods show that the vulnerability to consequential steam generator tube rupture depends on plant-specific design and operations.

3. NUREG-2195 methods should be applied in plant-specific risk assessments. Such evaluations should explore opportunities for mitigation strategies to reduce the potential for bypass accidents with offsite release of radionuclides from the reactor coolant system.

BACKGROUND

Since publication of the Reactor Safety Study in 1975, it has been understood that accidents involving release of radionuclides through paths that bypass containment pose significant risk to the public health and safety. In bypass accidents, there is no mitigation of radionuclide release by natural and engineered processes in the reactor containment. Bypass accidents may be risk-important even though they might not be important to core damage frequency. This early result was confirmed in a subsequent NRC study of representative power plants (NUREG-1150).

Initial concerns over bypass accidents focused on accidents initiated by a steam generator tube rupture, with additional failures that led to core degradation. Interfacing systems loss-of-coolant accidents were later recognized as other potential bypass accidents. Continuing development of the understanding of severe accidents raised the possibility that natural convection of steam and hydrogen through a degrading reactor core could impose heat loads threatening the integrity of steam generator tubes during accidents important to core damage frequency such as those involving a station blackout (SBO). That is, core damage accidents initiated by other means could evolve to become containment bypass accidents.

Two bounding cases were identified. In one case, loop seals of collected water within the reactor coolant system (RCS) could open allowing a system-wide, vigorous, circulation of hot gases through the steam generator. In the second, the loop seals are intact and circulation is from the core into the steam generators and back to the core along the hot leg. Simulant tests by Westinghouse confirmed that this counter-current natural circulation could develop. Of course, an intermediate case in which loop seals could clear and then reform can be imagined, but it is usually assumed that the threat of evolution to a bypass accident in this case is intermediate between the two bounding cases. If tube failures occur, these scenarios are termed consequential steam generator tube rupture (C-SGTR) accidents.

Further development of accident progression modeling has indicated that natural circulation of coolant through a degrading core could pose threats to the RCS at locations other than the steam generator tubes. Failures by natural circulation heating at vulnerable RCS locations, such as the hot leg or pressurizer surge line, would lead to a pathway for radionuclide release that opens into the containment where the radionuclides would be mitigated by natural and engineered processes. These RCS piping failures prior to steam generator tube damage are a persistently predicted feature of accident analyses without interventions of accident management efforts such as timely depressurization or alignment of FLEX makeup and cooling options.

In December 2009, the Office of Nuclear Reactor Regulation issued a user need requesting methods to facilitate review of risk assessments involving C-SGTR events. In particular, this user need expressed concerns that prior studies focused solely on Westinghouse plant designs and may not be applicable to Combustion Engineering (CE) plants that typically have different steam generator and RCS geometries.[1]

DISCUSSION

The staff has completed an important effort to address the user need concerning C-SGTR phenomena. This effort includes computational fluid dynamic (CFD) assessments, more detailed structural analysis of RCS components, development of a simplified risk assessment method for predicting thermally-induced and pressure-induced C-SGTR phenomena, and consideration of mitigating actions. Based on results from these thermal-hydraulic and structural analyses, the staff developed and applied a method for assessing the probability of C-SGTR and large early release frequency (LERF) for two pressurized water reactor designs

[1]Babcock & Wilcox designs are less susceptible to C-SGTR phenomena than Westinghouse and CE plants with U-tube steam generators. The elevations and design of the hot legs and use of once-through steam generators in Babcock & Wilcox designs tend to preclude tube ruptures that could lead to bypass events.

with U-tube steam generators: Zion Nuclear Power Station, a four-loop Westinghouse plant, and Calvert Cliffs Nuclear Power Plant, a two-loop CE plant. This effort considered severe accidents and design basis accidents.

Thermal-Hydraulic Analyses

There is considerable uncertainty in predicting natural circulation of gases through the RCS and the resulting temperature distributions for components such as piping and steam generator tubes. Natural circulation is design dependent, and the driving forces are accident specific. To specify the driving forces for natural circulation, the staff selected two accidents that are often found to be important contributors to the frequency of core damage. Time-dependent temperatures and pressures were calculated for two types of SBO sequences: one with early failure of the turbine-driven auxiliary feedwater pump (short-term SBO) and one with late failure of the turbine-driven auxiliary feedwater pump after battery depletion (long-term SBO).

Modeling of natural circulation requires describing the flow of gases circulating from the degrading core through the hot leg to the steam generator plenum and into the steam generator tubes. System codes are limited in their ability to predict these complex flow patterns. Such phenomena are better simulated by defining flow parameters based on calculations performed with other models. For the C-SGTR evaluations, CFD calculations using the FLUENT code were performed to provide these parameters. The CFD model was benchmarked using data obtained from 1/7th scale tests (NUREG-1781) and then applied to full scale Westinghouse and CE designs (NUREG-1788 and NUREG-1922). Then, MELCOR model parameters were developed to obtain values consistent with FLUENT predictions. Sensitivity calculations were performed to assess the impact of analysis assumptions affecting the timing of tube failure and containment bypass.

The calculations show that fluid flow rates and temperatures are strongly affected by steam generator geometrical features. The CE design has a shallow inlet plenum for flow mixing and a smaller hot leg length-to-diameter ratio. These features increase the heat load imparted to steam generator tubes in the CE design compared to the Westinghouse design.

The NUREG-2195 thermal hydraulic analyses have advanced the state-of-the-art for evaluating C-SGTR phenomena. Although benchmarking FLUENT with available data provides some confidence in its application to other plant designs with different geometrical features, there remain some limitations and uncertainties.

- *Analyses are available for only two designs.* The staff conducted a survey of replacement steam generator designs in other plants. Although a limited number of plants responded (less than 10% of operating plants), results indicate that the geometries of replacement units are similar to those in the studied plants. When NUREG-2195 methods are applied to a specific plant, it is important to confirm that it can be represented by one of these geometries.

- *Uncertainties in predicting natural circulation phenomena.* Evaluations indicate that phenomena, such as loop seal clearing and primary system pressure relief operation, can significantly affect heat loads to RCS components. For example, loop seal clearing accelerates heat transfer by allowing high temperature gases to pass from the core to the steam generator tubes. Results for the selected CE plant were less sensitive to loop seal clearing effects. Because less mixing occurs in their plenum design, high tube temperatures are predicted even if the loop seal is not cleared. Evaluations for the

Westinghouse plant indicated that loop clearing would occur only for reactor coolant pump seal leak rates on the order of 400 gpm. Such conclusions, however, are plant- and event-specific. NUREG-2195 methods can be applied in plant-specific risk assessments to better understand the effects of loop seal clearing and primary pressure relief related to natural circulation phenomena.

Enhanced Structural Analysis

Using thermal analysis results for Zion, finite element structural analyses were completed using the ABAQUS code. Evaluations focused on components most susceptible to failures with sufficient leakage to depressurize the primary system, including the hot leg, the surge line, manways, pressurizer relief valves, instrumentation and sample lines, and selected welds. Two-dimensional axisymmetric and three-dimensional models addressed plant-specific features, such as nozzle geometries and configurations, boundary conditions, loading conditions, fabrication effects, stress-corrosion cracking mitigation, and other degraded conditions. Models were updated with new high temperature creep and tensile data for component materials. The new data were incorporated into a model for predicting the time of creep rupture failure for each component and to evaluate the associated sensitivity to loadings.

These structural analysis efforts have advanced the state-of-the-art for evaluating C-SGTR. Results emphasize the importance of plant-specific differences in component geometry and fluid flow rates. The higher gas temperatures predicted for steam generators with a shallow inlet plenum and primary systems with a shorter hot leg increase the potential for C-SGTR following a core damage event with high reactor vessel pressure, a dry steam generator, and a low secondary pressure. An SBO followed by loss of auxiliary feedwater could lead to such conditions.

It is impractical to apply a detailed state-of-the-art finite element method, such as the ABAQUS code, to calculate failures for all the RCS components and conditions of interest. Rather, a simpler software tool, C-SGTR Calculator, was developed. This tool uses plant-specific design information, material properties, recently-measured steam generator flaw data, and scenario-specific thermal hydraulic results. It includes correlations for predicting tube failure due to pressure at low temperatures, new Argonne National Laboratory correlations for predicting high temperature creep rupture of flawed tubes, and Electric Power Research Institute correlations for predicting hot leg and surge line failure times. Failure predictions using C-SGTR Calculator were found to be consistent with results from more detailed analyses using the ABAQUS code.

C-SGTR Calculator was used to make numerous exploratory calculations to understand the effect of pressure and temperature challenges to steam generator tubes with specified tube flaw sizes and distribution information. Calculations assessed the impact of assumptions such as critical failure area for primary system depressurization, equipment failures, and human errors for representative accident conditions. Results were distilled into tables that show the failure probabilities for steam generator tubes and other RCS components. These tables provide a method for considering C-SGTR in existing risk assessments.

This new tool shows promise. For this promise to be fulfilled, more clarification regarding its current treatment of uncertainties is required. In some cases, uncertainties are omitted because their consideration was beyond the limited scope of this effort. In other cases, such as operator action uncertainty or relief valve performance under severe accident conditions, uncertainties are quantified without any supporting data. The manner in which uncertainties are currently treated and the process for updating values used to quantify uncertainties should be clarified in guidance developed for using this method.

Simplified Method for Assessing C-SGTR Risk

The C-SGTR effort focused on calculating two parameters that could be used with existing risk assessments: the conditional probability of C-SGTR given an accident that challenges steam generator tubes and use of this probability in evaluating LERF. NUREG-2195 assumes LERF as the frequency of events with a significant fission product release that occurs before there is confidence that 95% of the population can be evacuated. In NUREG-2195 evaluations, LERF is estimated as the product of five factors:

- Frequency of accident sequences with potential for C-SGTR
- Conditional probability for a sufficient number of steam generator tubes to experience induced failure and result in significant release of radioactivity that bypasses the containment
- Conditional probability that subsequent failures of the RCS do not occur
- Failure of severe accident mitigation actions
- Probability that early effective evacuation is not successful

The staff's efforts focused on developing a simplified method that would allow the frequency of sequences related to the C-SGTR evaluation to be obtained for plants with existing Level 1 probabilistic risk assessments (PRAs). The C-SGTR bypass frequency can be adjusted with a suitable multiplication factor or split fraction to account for the reduction in offsite release due to failures of alternate RCS components or other factors. This is similar to the method currently used for the risk significance determination process.

A major conclusion from this study is that the contribution of C-SGTR to containment bypass frequency for the CE plant was approximately a factor of ten higher than for the Westinghouse plant. The containment bypass frequency for the Westinghouse plant is estimated to be less than $1x10^{-6}$/year; for the CE plant, it is estimated to be as high as $1x10^{-5}$/year and the largest contributor to LERF. Results indicate that the conditional probability of C-SGTR is driven by the critical tube leak area needed to depressurize the primary system and the magnitude of the predicted temperature difference between hot leg and steam generator tubes (when temperatures are in the 600-800°C creep rupture range). The study acknowledged that a cleared loop seal could increase the conditional probability for C-SGTRs in the Westinghouse plant. For the CE plant, multiple tube failures are predicted in cases without loop seal clearing. Hence, loop seal clearing uncertainties are less important for the CE design.

We caution that existing PRAs are not constructed to identify all scenarios that are vulnerable to C-SGTR events. NUREG-2195 provides an approach that could be followed to identify C-SGTR sequences in existing Level 1 PRAs. Guidance for using NUREG-2195 methods should emphasize uncertainties associated with the use of current PRAs and expand upon the approach provided in NUREG-2195 to ensure that new Level 1 PRAs capture all such scenarios.

SUMMARY

We recommend issuance of NUREG-2195. Methods documented in this report have advanced the state-of-the-art for evaluating C-SGTR. Results provide significant insights regarding the importance of plant-specific design and operational features. Predictions using these methods indicate that the contribution of C-SGTR to containment bypass frequency for the selected CE plant is approximately a factor of ten higher than for the selected Westinghouse plant. Although the contribution of C-SGTR to containment bypass is small for the Westinghouse plant, its value could be higher because of uncertainties in predicting the potential for cleared loop seals. Forthcoming staff guidance is essential for using methods described in NUREG-2195 and for identifying associated limitations and uncertainties of these methods.

Work that the staff completed has identified potential vulnerabilities of some reactor designs to phenomena that can pose high risk. High-risk vulnerabilities merit attention, and staff work suggests that other areas of high risk might be revealed as better understandings of loop seal clearing and pressure relief effects are developed. The staff's work also suggests that mitigation strategies including those made possible by FLEX equipment could be developed to reduce the possibility that accident scenarios evolve to become bypass accidents with unmitigated release of radionuclides from the RCS. The methods suggested in NUREG-2195 should be applied in plant-specific risk assessments to better understand how these risk management options can be implemented most effectively.

Sincerely,

/RA/

Dennis C. Bley
Chairman

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