

SNUPPS

Standardized Nuclear Unit
Power Plant System

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SLNRC 86-05 FILE: 0278
SUBJ: Steam Generator Tube Rupture
Analysis - SNUPPS

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

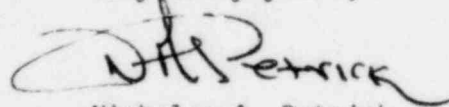
Docket Nos.: STN 50-482 and STN 50-483

- References:
1. NRC letter (B. Youngblood) to Union Electric Company (D. Schnell) dated March 13, 1986: Review of Steam Generator Tube Rupture Analysis
 2. NRC letter (B. Youngblood) to Kansas Gas & Electric Company (G. Koester) dated March 13, 1986: Review of Steam Generator Tube Rupture Analysis

Dear Mr. Denton:

The referenced letters requested that additional information be provided in support of the NRC review of the Steam Generator Tube Rupture Analysis for the SNUPPS plants - Callaway Plant Unit No. 1 and Wolf Creek Generating Station Unit No. 1. Enclosed are responses to the NRC staff questions.

Very truly yours,



Nicholas A. Petrick

MHF/dck/4a14

Enclosure

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RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION
SNUPPS STEAM GENERATOR TUBE RUPTURE ANALYSIS
CALLAWAY PLANT AND WOLF CREEK GENERATING STATION

QUESTION 1:

The SNUPPS analysis for the SGTR maximum overflow case states that at the time of break flow termination, the steam volume below the outlet nozzle is very small. Thus, the margin to overflow for this case is minimal, and a slight change in assumptions or calculational results could result in overflow. As an example, the SNUPPS analysis apparently assumes reactor trip at 100% power. This assumption may not be the most conservative from a standpoint of margin to overflow and is also probably not realistic when compared to the Ginna SGTR event. A more realistic scenario may involve turbine runback to some lower power followed by overtemperature delta T trip. At lower power levels the steam generator should have a larger liquid inventory because of reduced void fraction, assuming the SG level remains constant. Thus, starting maximum auxiliary feedwater flow at a lower power level may result in more rapid overflow. Discuss whether this scenario (i.e., lower void fraction) was considered in your analysis and what effect it would have on the margin to overflow.

RESPONSE:

The assumption of reactor trip at 100% power for the overflow case was found to be the most conservative. SNUPPS analyses included evaluation of turbine runback. Results of the analyses showed that the steam generator liquid volume was less than that obtained for the SGTR event with trip occurring at 100% power. This result was expected.

The overflow potential is dependent upon the amount of AFW added to the SG, the initial steam generator water mass, and the break flow.

AFW Flow. Auxiliary feedwater begins after reactor trip and continues until 16 minutes after the SGTR, for either case. Therefore, the time of reactor trip determines the duration of AFW flow. The analysis of an SGTR event with turbine runback showed that power would be reduced to 85% and that trip would occur as a result of low pressurizer pressure 540 seconds after the SGTR occurred. For the trip at 100% power case, trip occurs at 146 seconds after the SGTR. Since AFW begins after reactor trip, the trip at 100% power case initiates AFW 394 seconds earlier. This results in approximately 40,000 lbm less water for the turbine runback case (the AFW flow rate is approximately 100 lbm/second).

Initial SG Mass. The steam generator mass is approximately 7,000 lbs. greater for a turbine runback to 85% power at reactor trip than for the trip at 100% power case, as additional feedwater is added to maintain SG level at the lower power. This increase in initial SG mass for the turbine runback case is less significant than the reduction in SG mass due to the delay in AFW addition.

Break Flow. The analyses assumed that the SG level is maintained prior to reactor trip. Therefore, the break flow contribution to SG overflow depends on the time of reactor trip. The turbine runback case has a later trip and therefore will have less SG total mass due to the break flow.

Results of this analysis showed that the steam generator mass at AFW termination was approximately 25% less in the turbine runback case than that obtained for the SGTR event with trip occurring at 100% power.

Realistically, given an SGTR with turbine runback and no automatic trip expected for approximately 540 seconds, it is reasonable to expect that the operator would manually trip the reactor and terminate AFW flow well before the automatic trip occurs. This manual action by the operator would result in an even lower steam generator mass.

Thus, for the SNUPPS analysis, where an assumed failed AFW controller maximizes AFW flow, consideration of turbine runback to a lower power in the SGTR analysis was shown to reduce the potential for overfill.

QUESTION 2:

Explain the basis for the large difference for reactor trip time between the "failed open AFW control valve" case and the "stuck open ARV" case and the effect of these assumptions on the analysis results.

RESPONSE:

The reactor trip times are different for the "failed open AFW control valve" case and the "stuck open ARV" case because different assumptions were applied to each case.

Reactor trip occurs automatically as a result of overtemperature ΔT ($OT\Delta T$) in both cases analyzed. The rapid RCS depressurization caused by the SGTR reduces the $OT\Delta T$ setpoint to the point of reactor trip. The time of reactor trip is dependent upon the rate of depressurization (break flow rate), the initial RCS pressure and the $OT\Delta T$ setpoint parameters.

For the "failed open AFW control valve" case, the initial conditions and input parameters were chosen to maximize the potential for overfill. As discussed in Table 3-1 of the SNUPPS SGTR submittal, a cold leg break was chosen to maximize the total leaked reactor coolant, and a low initial RCS pressure (2220 psia) and $OT\Delta T$ setpoint were chosen to cause an early reactor trip time, thus maximizing the duration of the AFW flow to the faulted SG.

For the "stuck open ARV" case, the initial conditions and input parameters were chosen to maximize the offsite dose. As discussed in Table 3-2 of the SNUPPS SGTR submittal, a hot leg break was chosen to maximize the flashed fraction of leaked reactor coolant, a high initial RCS pressure (2280 psia) was chosen to maximize leakage flow and therefore flashed coolant and a high $OT\Delta T$ setpoint was chosen to maximize the ΔT at reactor trip and thus the flashed fraction. As described in Appendix F, maximizing the flashed fraction of leaked reactor coolant maximizes the offsite iodine dose.

The differences in these assumptions result in an early reactor trip time (146 seconds) for the "failed open AFW control valve" case and a late reactor trip time (503 seconds) for the "stuck open ARV" case. Each reactor trip time is conservative.

QUESTION 3:

The "stuck open ARV" case assumes that the atmospheric relief valve (ARV) is isolated in 20 minutes by manually closing the ARV block valve. State how this time period was established and whether it is realistic considering that this operation would be performed in a location subject to adverse conditions including high temperature, radiation and noise.

RESPONSE:

In the "stuck open ARV" case, the atmospheric relief valve is isolated in 20 minutes by manually closing the ARV block valve. The isolation time is based on an expected action time of 4 to 8 minutes plus additional margin added for conservatism to maximize offsite dose in the analysis.

The bases for the 4 to 8 minutes expected action time are:

1. Physical distance and location,
2. Operator interviews and experience,
3. Actual tests in manning the Auxiliary Shutdown Panel at Callaway, and
4. Actual results from closing an ARV block valve at Callaway.

During an emergency situation, control room personnel dispatched to operate the SG ARV block valves would walk from the Control Room through the Secondary Alarm Station (SAS) and into the Control Room Filtration Room in the Auxiliary Building. Once in the Auxiliary Building, the operator would have a direct route into the Steam Tunnel as shown in Figures 1.2-13 and 1.2-25 of the SNUPPS FSAR. The total distance is approximately 180 feet. The expected travel time is 1 to 3 minutes.

Callaway equipment operators indicated that, once in the Steam Tunnel, it would require personnel 3 to 5 minutes to identify and operate the SG ARV block valves. As a result, the total isolation time is estimated to be between 4 and 8 minutes.

The validity of this estimate is supported by an exercise to evacuate the Control Room and man the Auxiliary Shutdown Panel at Callaway during startup testing. In this exercise, which was observed by the NRC, personnel were able to evacuate the Control Room and operate the Auxiliary Shutdown Panel within 5 minutes. Given the proximity of the Auxiliary Shutdown Panel to the Steam Tunnel (see FSAR Figure 1.2-24), this exercise provides a good measure of ARV block valve actuation time.

In October of 1984, an operator was dispatched from the Control Room to close an ARV block valve at Callaway. This action was completed in 16 minutes, including time (10 minutes) to obtain a health physicist to badge the operator. Since that time, dosimetry has been made available in the control room; and little time would be required for the operator to obtain a radiation badge. The operator was not hampered by the high temperatures or the noise associated with steam escaping through the ARV.

Callaway operators have demonstrated that the ARV can be closed in the aforementioned time frame given the "adverse" temperature and noise conditions found in the Steam Tunnel during power operations. These same

conditions would be applicable to a steam generator tube rupture event. Furthermore, the radiation levels outside the steam lines are minimal given the low concentrations and character of the radioisotopes (low gamma activity) associated with an SGTR event.

QUESTION 4:

Appendix E "Bases for ARV Technical Specification" states: "An ARV is considered operable if the block valve is closed solely because of leakage." The SGTR analysis assumes that the operator initiates RCS cooldown in less than 30 minutes by opening the intact SG ARVs. Since the operator may have to open the ARV block valves manually if the above Technical Specification is implemented, demonstrate that this can be accomplished within the stated time frame considering the concerns regarding this operation expressed in Question 3.

RESPONSE:

For the overfill case, it is stated that the operators complete identification and isolation of the faulted SG within 16 minutes and initiate cooldown within 24 minutes of the SGTR. The identification time period for an SGTR has been conservatively estimated to be 12 minutes based on a 10 minute delay time and 2 minutes for the identification procedure. This conservative estimate is based on simulator experience and ANS 58.8 criteria. Therefore, isolation of the faulted SG by the control room operators is accomplished in 4 minutes (16 minutes for the total time period less 12 minutes for SGTR identification and initiation of operator action). Because initiation of opening the SG ARV block valve can proceed after the SGTR identification time period (12 minutes) is complete, an equipment operator would have 12 additional minutes to open an intact SG ARV block valve and commence cooldown.

This assumed 12 minute interval is conservative relative to the 4 to 8 minutes expected for travel to the block valve location and valve manipulation and is therefore consistent with the assumptions in the analysis.

QUESTION 5:

In your analysis, you assumed that the fission products released to the intact steam generators were not released to the environment. Provide an analysis demonstrating that the fission products released to the intact steam generators will be retained in the steam generators during the cooldown phase.

RESPONSE:

Appendix F of the SNUPPS SGTR report describes the calculation of doses resulting from radioactivity releases from both the intact and faulted steam generators. A total of 1 gpm reactor coolant leakage was assumed into the secondary side of the intact steam generators and was assumed to continue throughout the accident. As depicted in Table F-2, steam is released to the atmosphere from the intact steam generators during the time

intervals 1, 2, 4, 7 and 8 (cooldown phases occur during time intervals 4 and 7). This released steam is assumed to contain an iodine concentration equal to 1% of the water concentration on a mass basis (a partition factor of 100). Noble gas releases were calculated to be equal to 100% of the noble gas contained in the 1 gpm leakage flow to the secondary side of the intact steam generators.