SNUPPS

Standardized Nuclear Unit Power Plant System

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Executive Director

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SUBJ: Steam Generator Tube Rupture Event

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

Dear Mr. Denton:

This letter is in response to three questions forwarded informally to SNUPPS by the NRC concerning the analysis of a steam generator tube rupture (SGTR) event, contained in the SNUPPS FSAR. SNUPPS is in the process of reanalyzing the SGTR taking into account operator action times from: actual SGTR events, ANS Draft Standard 58.8, and plant simulators. While the analyses are not yet complete, they are far enough advanced to provide preliminary responses to the NRC's questions. These responses are considered adequate to justify interim operation of the Callaway and Wolf Creek plants. SNUPPS will continue to refine its analyses and will provide a full report to the NRC by June 30, 1984. It is our understanding that a group of utilities with Westinghouse plants are also reexamining SGTR events and are working to about the same schedule as SNUPPS.

Question one requests the applicant to submit an evaluation of operator actions necessary to effect pressure equalization and a conservative time estimate for each action as well as an initial delay time. Based on actual SGTR events, simulator data, and ANS Draft Standard 58.8, it is concluded that the following response times, measured from occurrence of a SGTR, are reasonable and conservative for the SNUPPS plants.

- Isolate auxiliary feedwater to faulted SG	16 min
- Initiate RCS cooldown by depressurizing intact SGs	32 min
- Equalize pressures of RCS and faulted SG	40 min
- Terminate safety injection	45 min

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Except for isolation of auxiliary feedwater to the faulted SG, these times are consistent with the guidance of ANS Draft Standard 58.8 for a condition IV event. The time of isolation of AFW is consistent with ANS Draft Standard 58.8 for a condition III event and has been used because it is supported by simulator data and because control of SG water level is a common operation, aided by high and low level alarms.

The justification for the time to effect pressure equalization being shorter than for the Ginna event is as follows. A major reason for the long time at Ginna was that a pressurizer PORV stuck open, resulting in lower RCS pressure than desired. The SNUPPS plants have fully qualified pressurizer PORVs with fully qualified block valves. Further explanation of the timing of the Ginna SGTR is that the Ginna plant has a reactor vessel upper head temperature approximately equal to the hot leg temperature. Depressurization of the RCS resulted in formation of a steam bubble in the vessel head. The SNUPPS plants are designed to have cold leg temperature in the upper head and thus are less prone to formation of a steam bubble. In the Ginna event the RCS was repressurized by prolonged operation of safety injection. The Ginna operators were apparently reluctant to terminate safety injection while there was evidence of steam voids in the RCS and it is our understanding that the emergency procedures in effect at that time were not explicit about termination of SI with voids in the RCS. The emergency procedures have now been clarified on that point and, as noted, the SNUPPS plants are less likely to have voids in the RCS. Furthermore, it is probable that offsite power would be retained. The SGTR emergency response procedure has now been clarified to keep reactor coolant pumps in operation, which would reduce further the possibility of incurring steam voids in the RCS.

Question two is whether, as a result of varying parameters in the SGTR analysis, liquid could enter the main steam lines. The results of analyses performed by SNUPPS show that, with one exception, the combinations of operator response times listed above and various postulated single failures result in no liquid entering the steam lines. The exception is a failure in the open position of the auxiliary feedwater valve to the faulted steam generator. It is likely that, in this case, the operators would respond more rapidly (i.e., within 10 min), on the basis of high level alarms, to isolate auxiliary feedwater to the faulted SG. If the auxiliary feedwater is isolated within 10 minutes and the other response times are as listed above, the accident would be terminated without water entering the steam line.

The second part of question two is what would be the effects on the integrity of the steam piping and supports if liquid should enter the steam lines. The answer is that the integrity of the piping and supports would not be affected. The steam lines and supports of the SNUPPS plants have been designed for the case where the steamlines are filled with water. The possibility of damaging water hammer is extremely remote. Liquid would enter the steam line slowly. Furthermore the steam and water in the steam line would be at nearly the same temperature and, therefore, condensation shocks would not occur.

The third question pertains to the safety classification of components that are credited to mitigate the consequences of a SGTR. In the SNUPPS plants, all of these components are safety-related, are qualified to environmental and seismic conditions, and are powered by Class 1E power. These components include: the pressurizer PORVs, the SG atmospheric relief valves, all necessary auxiliary feedwater components, safety injection components, and the backup supply of auxiliary feedwater (essential service water system).

Based on the preliminary results presented herein, there is adequate assurance for safe interim operation of Callaway and Wolf Creek. Final results will be forwarded to the NRC by June 30, 1984, as stated above. Necessary FSAR revisions will follow thereafter.

Very truly yours,

N A Petrick

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