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October 14, 1983

Docket No. 50-244

LICENSEE: ROCHESTER GAS AND ELECTRIC CORPORATION

FACILITY: R. E. Ginna Nuclear Power Plant

SUBJECT: SUMMARY OF OCTOBER 5, 1983 MEETING TO DISCUSS THE INPO SIMULATION OF THE GINNA STEAM GENERATOR TUBE RUPTURE EVENT

On October 5, 1983 members of the NRC staff met with representatives of Rochester Gas and Electric Corporation (RG&E), Institute of Nuclear Power Operations (INPO), and Argonne National Laboratories (ANL) for the subject discussion. Using steam generator tube rupture (SGTR) information obtained from RG&E, INPO modeled the event using the RETRAN-02 computer code. INPO will provide the program to the NRC and ANL. It will be used to study the thermal-hydraulic response of the system during the SGTR and the implications regarding pressurized thermal shock.

The attendance list (Enclosure 1) and agenda (Enclosure 2) are enclosed. INPO provided an overview of how the SGTR event was modeled and the analysis of the results, (Enclosure 3) summarizes the general information provided. The answers to specific technical questions were provided by INPO.

ANL will be provided a copy of the program tape and the data necessary to run the program within 3 weeks. Adapting the program to another computer, running it, and analyzing the results is expected to take approximately 10 months.

Original signed by
George F. Dick, Jr., Project Manager
Operating Reactors Branch #5
Division of Licensing

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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George F. Dick, Jr., Project Manager
Operating Reactors Branch #5
Division of Licensing

ATTENDANCE LIST
OCTOBER 5, 1983 MEETING

NAME	ORGANIZATION
Robert Eliasz	RG&E
Robert Mecredy	RG&E
Roger Wyrick	INPO
Gary Fader	INPO
William Brown	INPO
Edward Winkler	INPO
Jack Tessier	ANL
Tom Wie	ANL
Jack Guttmann	NRC
George Dick	NRC

Enclosure 1

AGENDA

for

Meeting with INPO, RG&E, NRC, and ANL

on

RETRAN Analysis of Ginna Steam Generator Tube Rupture Event

8:30	Welcome	G. Fader
8:45	Introduction	R. Mecredy
9:00	NRC Analysis Goals & Objectives	J. Guttman
9:30	INPO/RG&E RETRAN Model	R. Wyrick
	- Objectives of INPO Analysis	
	- Observations and Results	
	- Important Physical Phenomena	
	- Modeling Assumptions and Limitations	
	- Conclusions and Suggestions	
12:00	Lunch	
1:00	Additional Technical Discussions	All
3:00	Adjourn	All

Enclosure 2

THERMAL-HYDRAULIC ANALYSIS OF THE GINNA STEAM GENERATOR TUBE RUPTURE EVENT USING RETRAN-02

R. K. WYRICK, E. N. WINKLER, AND W. W. BROWN

Institute of Nuclear Power Operations
Atlanta, Georgia

INTRODUCTION

A detailed analysis of the thermal-hydraulic response of the R. E. Ginna nuclear power plant during a major steam generator tube rupture event of January 25, 1982, was performed. The event illustrated a number of thermal-hydraulic phenomena and operational problems that can occur during tube rupture transients. For this analysis, a computer simulation of the Ginna event was performed using the RETRAN-02 computer code (Ref. 1) and the results compared to the available plant data and sequence of events. The objectives of this analysis included the following:

- a thorough understanding of the thermal-hydraulic response of a nuclear power plant during a steam generator tube rupture transient
- better understanding of the sequence of events and physical phenomena that occurred during the Ginna event than was available from the plant data
- better understanding of various procedural actions and their corresponding affect on the physical response of a nuclear power plant during steam generator tube rupture transients.

GINNA EVENT OVERVIEW

The R. E. Ginna nuclear plant, is a Westinghouse-designed two-loop pressurized water reactor (PWR) that has been in operation since 1970. The plant full-power output is 1520 Mwt. On January 25, 1982, a single tube ruptured in the "B" steam generator (SG) at Ginna, initiating a complex plant transient,

lasting approximately 33 hours, until cold shutdown conditions were reached. Salient occurrences and phenomena during the event included the following:

- rapid reactor coolant system (RCS) depressurization, reactor trip on low pressure, and initiation of safety injection (SI)
- natural circulation cooling of the RCS due to procedural requirements to stop reactor coolant pumps
- formation of a steam region in the reactor vessel upper head during RCS depressurization by opening of a pressurizer PORV
- overfilling of the "B" SG and flooding of the steam line with water
- pressurization of the damaged "B" SG and steam line, causing opening of an SG safety valve
- liquid releases through the "B" SG safety valve, as indicated by radiological analysis of site samples
- higher temperatures in the "B" SG and steam line than in the RCS during plant cooldown.

ANALYSIS

The thermal-hydraulic response of the Ginna steam generator tube rupture event was analyzed by simulating the transient with the RETRAN-02 computer code and comparing the calculated results with the available plant data obtained from the plant computer and operator logs. The RETRAN-02 model (see Figure 1) consisted of 44 fluid volumes, 61 junctions, and 24 heat conductors. Initial conditions from the event were used first to develop a steady state case to establish the numerical stability of the model. The transient was initiated by causing a break flow area of 0.0066 ft^2 in a "B" SG tube.

The tube rupture flow rate was calculated using the extended Henry-Fauske critical flow correlation (Ref. 2) available in the RETRAN code. A discharge coefficient for the break of 0.475 was determined by a series of parametric studies in which the discharge coefficient was varied until the calculated pressurizer and RCS pressures reasonably matched the available pressure data.

Twelve flow boundary conditions were used during the transient calculation. Eight were RETRAN fill junction boundary conditions for "A" and "B" steam flow to the turbine/condenser, "A" and "B" main feedwater flow, "A" and "B" auxiliary

feedwater flow, RCS charging pump flow, and RCS letdown flow. These flows were determined from a combination of known valve and pump actions obtained from the plant computer and operator logs, and iterative variation of the flow rate within design limits until calculated plant transient results were in reasonable agreement with the available plant data. The plant data included RCS and steam generator pressures, pressurizer and steam generator water levels, reactor vessel upper head and core exit fluid temperatures, and RCS cold leg temperatures.

The other four boundary conditions were safety injection pump flow, pressurizer PORV flow, "A" SG atmospheric steam dump valve flow, and "B" SG safety valve flow. Safety injection flow was calculated using the RETRAN control logic from pump performance data for flow versus RCS pressure. The pressurizer PORV, the "A" SG atmospheric steam dump valve, and the "B" SG safety valve were modeled by connecting the downstream side of the valves to a large volume at atmospheric pressure. Flow through the valves was calculated using the extended Henry-Fauske (Ref. 2) and Moody (Ref. 3) critical flow correlations available in the RETRAN code.

Reactor Vessel Upper Head Modeling

Flashing of fluid in the reactor vessel upper head region was suspected to have occurred during the event. This was based on temperature indications from thermocouples located in the lower portion of the upper head and pressurizer level data which showed a rapid increase during RCS depressurization, when a pressurizer PORV was opened. The calculation of steam formation, the size of the steam bubble, and its rate of collapse required a model capable of a reasonable prediction of the temperature distribution in the upper head fluid.

The actual reactor vessel upper head geometry and flow distribution is complex, involving flow paths connecting with the outlet plenum region through 33 control rod guide tubes, and with the downcomer region through flow holes in the upper support plate. A relatively simple RETRAN model was used to allow a two region axial temperature distribution in the upper head. (See Figure 1) The final model was developed by a series of transient calculations in which the flow rate (i.e. the junction loss coefficients) entering the upper head and the relative sizes of the two upper head volumes were varied to determine an upper head enthalpy distribution required to predict the available upper head temperature and pressurizer level data. The RETRAN non-equilibrium thermodynamic "pressurizer" option was used to facilitate modeling water level and the subsequent collapse of the steam bubble in the upper head.

"B" Steam Line Modeling

During the event the "B" SG water level increased to the top of the SG and overflowed into the "B" steam line. The "B" steam line eventually filled completely with water. The SG and steam line model included the use of the non-equilibrium thermodynamic "pressurizer" volume option in the SG and in the steam line to improve the modeling of non-equilibrium effects during the filling of the steam line.

The spray junction option and the steam-water heat transfer coefficient of the "pressurizer" model in the steam line were varied in a series of transient calculations to provide condensation and compression rates necessary to match the available "B" SG pressure data. Once the pressure data was matched the model provided a reasonable estimate of the water level transient in the steam line.

RESULTS

Selected calculated transient results from the analysis and comparison to plant data, where available, are presented in Figures 2-6. Figure 2 shows calculated flow rates for the tube rupture flow, safety injection pump flow, and charging pump flow. Figure 3 shows a comparison of calculated RCS and steam generator pressures with the available plant data. Figure 4 shows calculated temperatures in the reactor vessel upper head and at the core exit. The upper head temperatures include the liquid temperature in the bottom region of the upper head (RETRAN Volume 20), and the liquid and steam temperatures in the top region of the upper head (RETRAN Volume 19). Available data from thermocouples located in the bottom region of the upper head (near the interface of RETRAN Volumes 19 and 20) and at the core exit are also shown in Figure 4. Figure 5 shows the calculated reactor vessel upper head steam volume. Figure 6 shows the calculated RCS cold leg temperatures and their comparison to the available cold leg temperature data.

OBSERVATIONS AND CONCLUSIONS

Principal observations and conclusions of the analysis are briefly summarized below for one hour and twenty minutes of the transient, during which time most of the significant thermal-hydraulic phenomena occurred. Further details regarding the Ginna event and its thermal-hydraulic response can be found in References 4, 5, 6, and 7. Elapsed event time is expressed as hours:minutes:seconds.

1. The maximum primary-to-secondary flow rate through the ruptured tube occurred shortly after the rupture and was calculated to be 632 gpm (62 lbm/sec.).

2. After the reactor coolant pumps were stopped at 0:04:10, flow into the reactor vessel upper head was calculated to decrease to about 1 percent of the normal flow rate. As a result of this flow reduction, the vessel upper head fluid cooled more slowly during plant cooldown than the remaining RCS fluid and became the hottest region in the RCS. This condition lasted approximately 2 hours, until a reactor coolant pump was restarted at 1:57:00.

3. When a pressurizer PORV was opened at 0:42:30, the hottest fluid region in the RCS was at the top of the reactor vessel upper head. The fluid temperature there was calculated to be about 20 ^{o}F hotter than the fluid in the bottom region of the upper head (where three thermocouples were located) and about 100 ^{o}F hotter than the core exit fluid. The calculated subcooling margin in the hottest fluid region was only 4 ^{o}F at the time the PORV was opened.

4. Formation of steam was calculated to have occurred in the reactor vessel upper head region during the initial pressure decrease following the tube rupture and during the depressurization associated with the pressurizer PORV openings. The calculation indicated that 7.5 ft^3 of steam formed during the initial pressure decrease and approximately 200 ft^3 of steam formed during the pressurizer PORV openings.

5. Based on this analysis steam did not form in the RCS hot legs or steam generator U-tubes at any time during the transient.

6. Complete isolation of steam flow from the affected "B" steam generator appears to have occurred about 6 minutes after closure of the "B" MSIV at 0:15:00. It is likely that flow from the "B" steam generator to the turbine-driven auxiliary feedwater pump was not isolated until the pump was stopped at 0:21:00.

7. Based on the calculation, flow through the ruptured tube caused the water level to increase in the "B" steam generator, causing compression and superheating of the steam regions at the top of the "B" steam generator and in the "B" steam line (after the steam line was completely isolated).

8. After 0:42:30, when steam formed in the reactor vessel head, the system pressure was influenced by three effective "pressurizers." These included the real pressurizer, the steam/water interaction in the reactor vessel upper head, and

the steam/water interaction in the "B" steam line.

9. The calculation indicated that the "B" steam generator began overflowing into the "B" steam line at about 0:43:10. The SG overfill caused a slight reduction in "B" SG pressure due to the condensing effect of subcooled fluid from the "B" SG on the superheated steam in the "B" steam line.

10. Based on this analysis, the "B" steam line header, where the SG safety valves are located, filled with water at about 0:47:30, prior to the first safety valve opening. Therefore, water releases occurred during all openings of the "B" SG safety valve during the event. The calculation indicated that the entire "B" steam line filled completely with water up to the MSIV at 0:58:28.

11. One of the four "B" SG safety valves opened several times during the transient. During all of the valve openings, the valve flow area was calculated to be less than the full-open flow area. This calculated result was consistent with the post-event examination of the valve, which also indicated that the valve never reached a full lift position.

12. Based on the calculation, the "B" safety valve opened and closed at least once before the "B" steam line filled solid with water. After the "B" steam line became water-solid at 0:58:38, the analysis indicates that the safety valve flow area increased and decreased corresponding to pressure changes in the steam line, but likely did not close completely during the two-hour and four-minute interval from 0:58:00 until 3:02:00. The inability of the valve to close can be explained by the continued flow through the ruptured tube (due to RCS safety injection and/or charging) into the water-solid "B" SG and steam line.

13. Based on the plant data, the stopping of safety injection at 1:12:00 caused RCS pressure to decrease from about 1370 psig to about 930 psig at 1:16:00. The "B" SG pressure correspondingly decreased from about 1055 psig to 840 psig. The RCS and "B" SG pressures decreased after stopping safety injection but did not equalize. "B" SG pressure remained 50-100 psi less than RCS pressure until 3:02:00. This was most unlikely due to continued leakage through the "B" SG safety valve during this time period.

14. Based on this analysis the RCS and "B" SG pressures would have equalized after safety injection had been stopped if charging flow had been reduced to the value of the letdown flow rate.

15. With reactor coolant pumps off and natural circulation flow nearly stopped in the "B" primary coolant loop, decreased thermal mixing of safety injection fluid in the cold leg significantly reduced the "B" cold leg fluid temperature. The analysis indicated that the net flow direction in the nearly stagnant "B" cold leg was directed toward the reactor vessel rather than toward the ruptured SG tube. The reactor vessel wall directly beneath the "B" inlet nozzle likely experienced much cooler water than other reactor vessel regions, creating a potential for thermal shock of the vessel wall. (Fracture analysis performed by Westinghouse indicated that no flaw propagation occurred during the Ginna event.)

16. Continued operation of safety injection and charging caused RCS presssure to remain higher than the safety valve opening setpoint of the damaged steam generator during most of the time between 0:00:00 and 1:12:00. Continued safety injection and charging caused the "B" steam generator to overfill and to flood the steam line, and caused "B" steam generator pressure to increase and a steam generator safety valve to open several times.

REFERENCES

1. "RETRAN-02, "A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM, May, 1981.

2. Henry, R.E. and Fauske, H. K. "The Two-Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices, and Short Tubes, "J. Heat Transfer, 93, 179-187, 1971

3. Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture, J. Heat Transfer, 87, 134-142, 1965

4. "Analysis of Steam Generator Tube Rupture Events at Oconee and Ginna," INPO Report 82-030, November, 1982

5. "Thermal-Hydraulic Analysis of Ginna Steam Generator Tube Rupture Event", INPO Report 83- , 1983

6. "Incident Evaluation, Ginna Steam Generator Tube Failure Incident, January 25, 1982, R. E. Ginna Nuclear Power Plant," Rochester Gas and Electric Company, April 12, 1982

7. "NRC Report on the January 25, 1982, Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," NUREG-0909, April 1982.

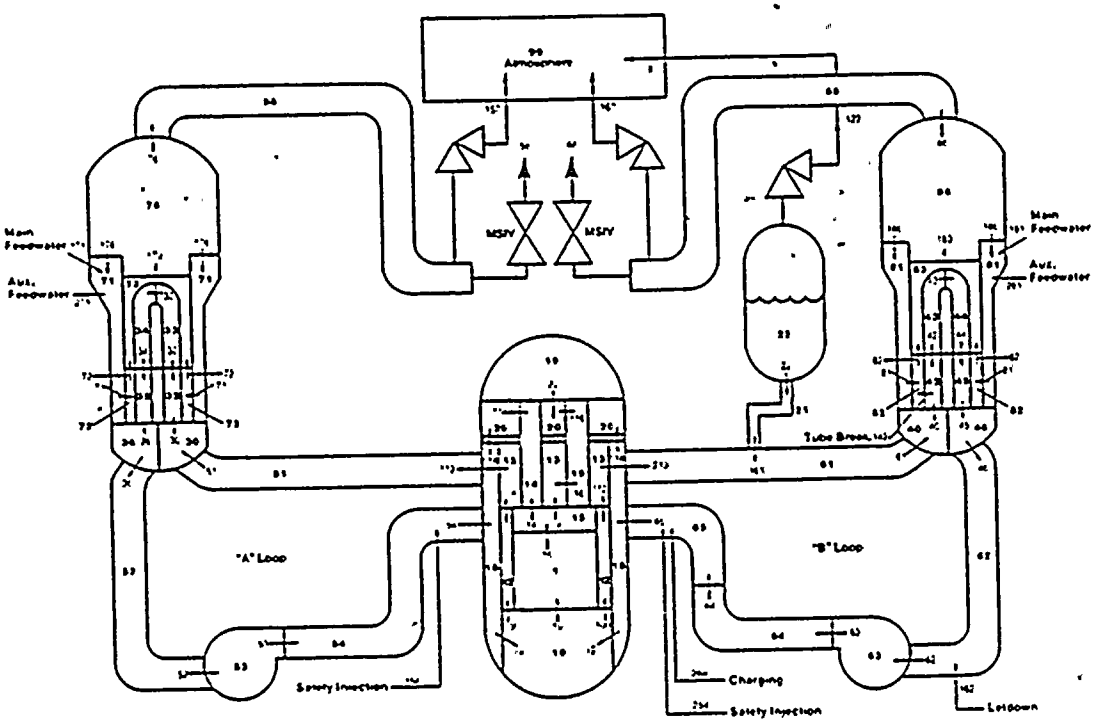

Figure 1. Retran Model Volumes and Junctions

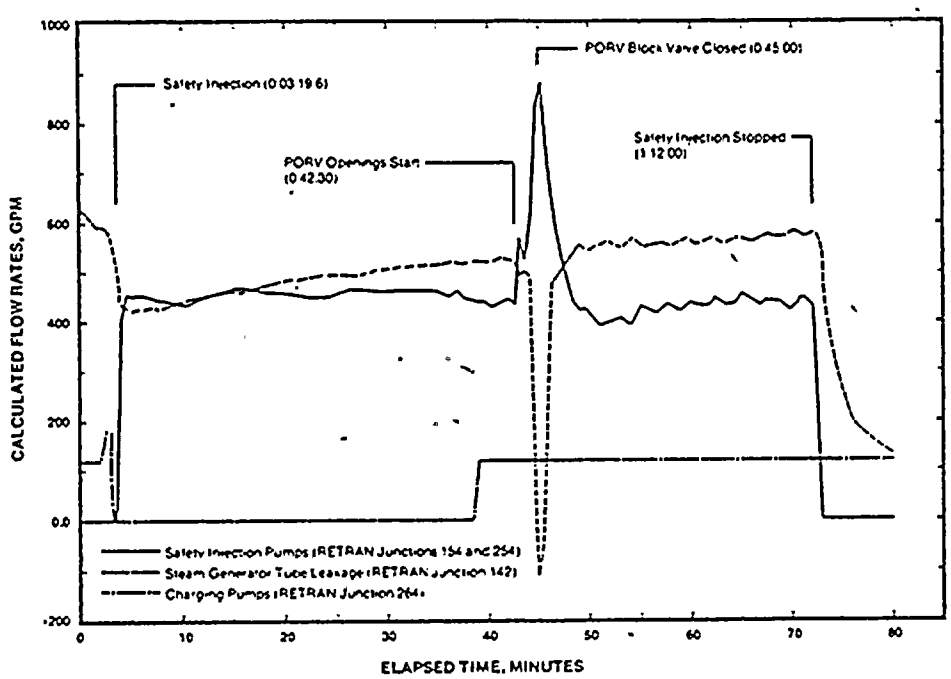

Figure 2. Steam Generator Tube Rupture, Safety Injection and Charging Flow Rates vs. Time

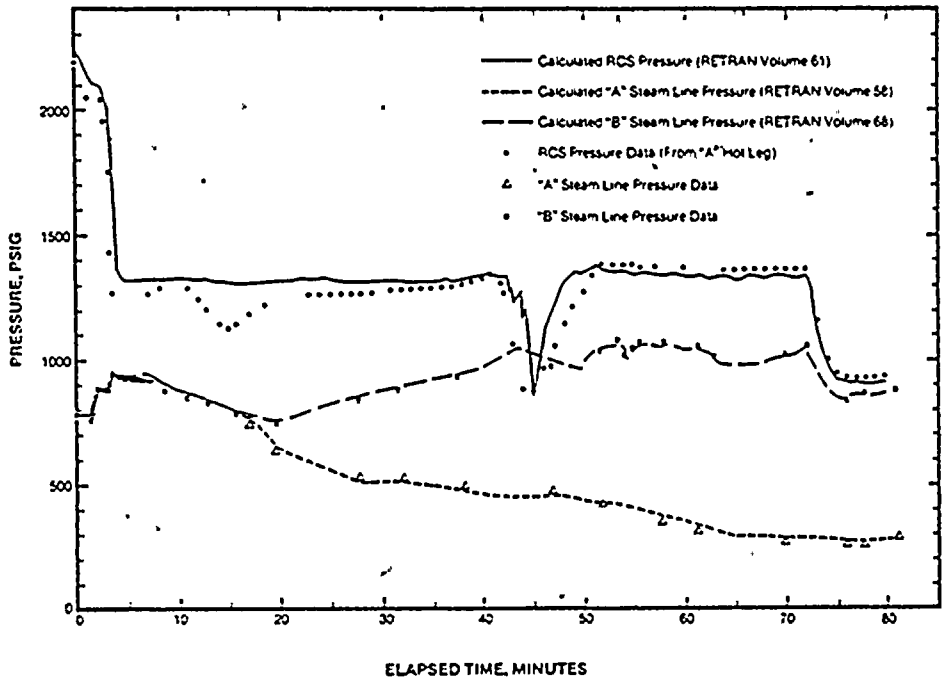

Figure 3. RCS and Steam Line Pressures vs. Time

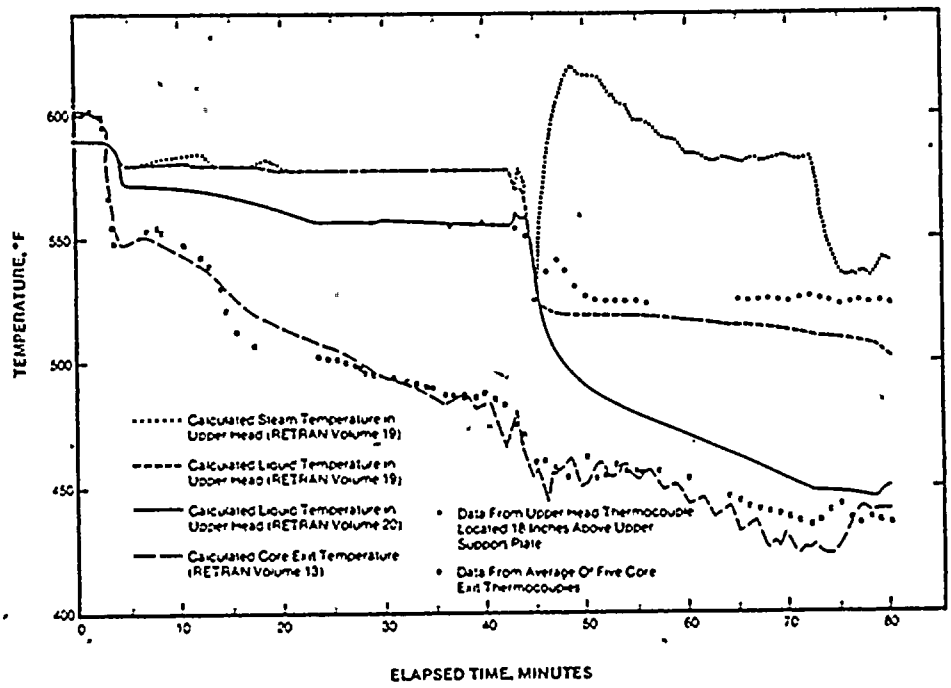

Figure 4. Reactor Vessel Upper Head and Core Exit Temperatures vs. Time

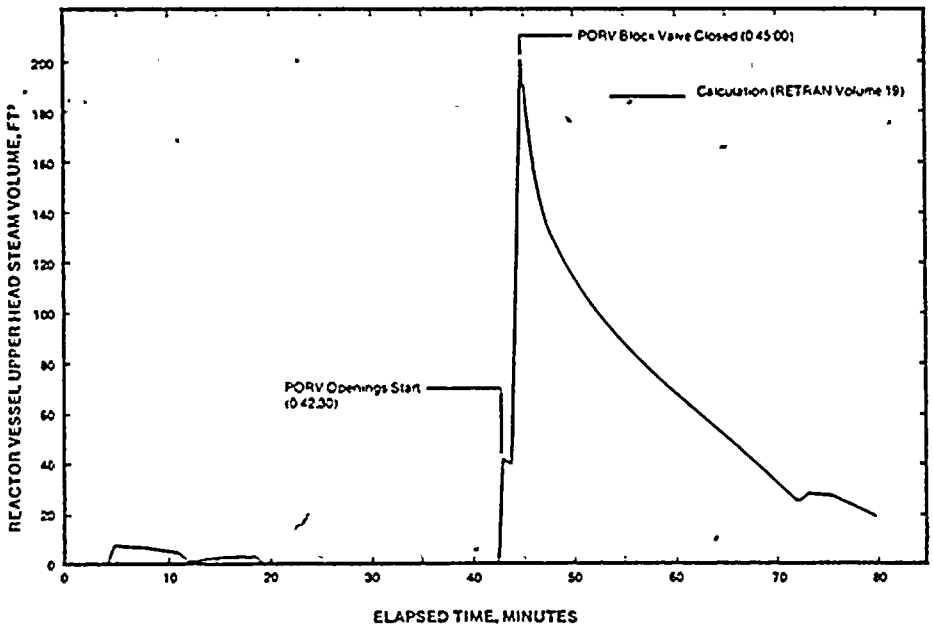

Figure 5. Reactor Vessel Upper Head Steam Volume vs. Time

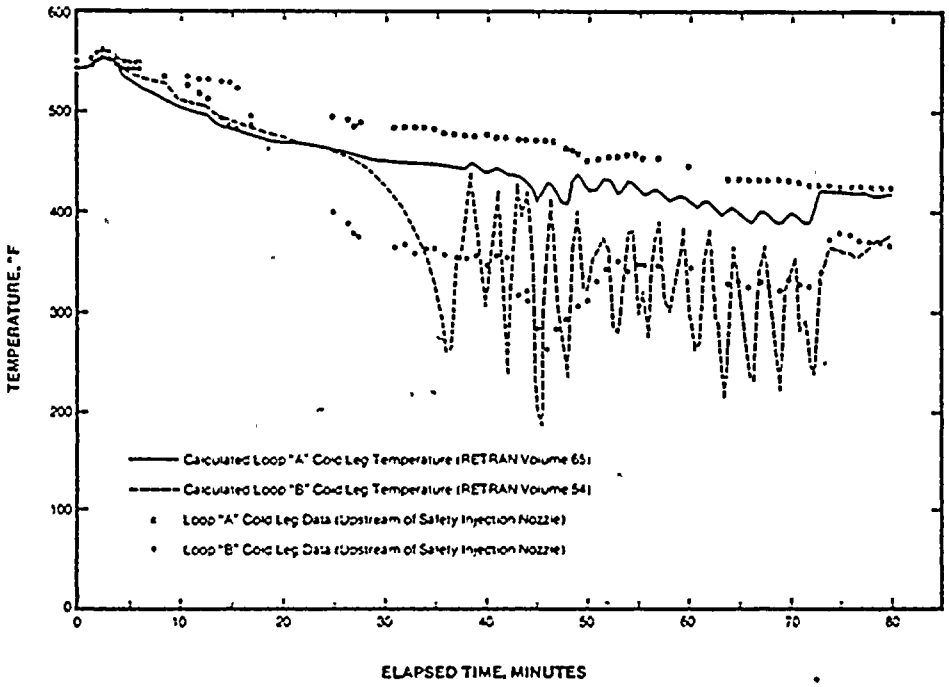

Figure 6. RCS Cold Leg Temperatures vs. Time