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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

The Safety Evaluation Report (NUREG-0916) related to the restart of Ginna after the steam generator tube rupture (SGTR) incident on January 25, 1982 and specifically license conditions 2.C(9)1 through 20 required that Rochester Gas and Electric Corporation (RG&E) address 20 long-term items. One of the license conditions, 2.C(9)7 required detailed thermal-hydraulic analysis of system behavior during the incident to verify phenomena including void formation.

By letter dated November 22, 1982 (Reference 1) RG&E provided sufficient information to the staff to evaluate the licensee's response to the staff's concerns.

2.0 BACKGROUND

Steam voids can form in part of the reactor coolant system any time the pressure falls below saturation for that part of the system. During the SGTR, it appeared to have occurred twice, first during the initial sharp depressurization following reactor trip and then during the second sharp depressurization when the PORV stuck open (Reference 2).

Despite formation of upper head voids there was never any indication that the core was not subcooled. Thus, effective cooling was always maintained and the existence of the bubble in the upper head posed no threat of core uncovery or inadequate core cooling. However, the actual extent and timing of void formation and condensation was unclear. The staff concluded that it was an important issue. Consequently the licensee agreed to perform, within six months, a detailed thermal-hydraulic computer analysis, of the reactor system behavior during the event, to verify the key thermal-hydraulic phenomena that may have taken place.

3.0 DISCUSSION

The licensee addressed the thermal-hydraulic analysis of the SGTR using the LOFTRAN computer program (Reference 3). The LOFTRAN computer program has been approved for use in Safety Analysis Reports (SAR) for analyses of Chapter 15 design basis events. These analyses include the steam generator tube rupture event. LOFTRAN was shown to produce conservative licensing evaluations by use of proper selection of input data and by use of the models employed in the LOFTRAN computer program. Since LOFTRAN is designed as a conservative licensing tool, it does not always exactly predict the sequence of a given accident. Several limitations were identified which were significant when applied to the Ginna event. These were considered by the licensee and a number of auxiliary calculations were presented as necessary to more accurately study the SGTR event. The purpose of this evaluation was to verify the thermal-hydraulic phenomena during the incident, including void formation.

4.0 EVALUATION

The parameters addressed for this evaluation were: (1) primary system pressure, (2) reactor coolant flow, (3) reactor coolant temperature, (4) pressurizer level, (5) break flow, (6) reactor coolant voiding, and (7) steam generator overfill. A long term recovery evaluation was also provided.

4.1 Pre-Trip System Response

The pre-trip system response analysis was performed using, as input, normalized core power and secondary pressure, as obtained from the plant data recorders, to evaluate the reactor coolant temperature and pressure response. Pressurizer pressure and level calculations agreed well with the plant data. It was also demonstrated that the pressure and level are significantly affected by the coolant temperature trends.

4.2 Post-Trip System Response

The post-trip system response analysis was performed using the recorded intact steam generator pressure as input to the LOFTRAN computer program. The LOFTRAN calculations were performed with artificial steaming of the generator in order to reproduce the recorded subcooling in the associated cold leg for the transient period from 7 to 16 minutes following the SGTR.

4.3 Reactor Coolant Pressure

A small void may have developed in the upper head region during the initial depressurization (4 to 5 minutes) although LOFTRAN did not predict flashing at this time. Following safety injection, LOFTRAN calculated a more rapid repressurization than was observed in the plant response. This has been attributed to the collapse of an upper head void during the actual event. The LOFTRAN calculation slightly overestimated the pressure response prior to the PORV cycling. An upper head void was generated during PORV cycling. Following isolation of the failed open PORV, the LOFTRAN calculation showed a more rapid repressurization than was experienced at the plant. This was attributed to the LOFTRAN limitation which inhibits refill of the upper head region void during natural circulation. The actual plant response, a slower repressurization, was attributed to at least partial refill of the upper head region.

4.4 Reactor Coolant Flow

The LOFTRAN calculation indicated that natural circulation through the intact loop was maintained between 3% and 4%, until reactor coolant pump (RCP) restart. Flow stagnation in the faulted loop was calculated to occur at about 45 minutes. The LOFTRAN calculation did not predict significant reverse flow through the faulted loop, however, the effect of the break flow model on the calculated loop flow was uncertain. An evaluation performed by the licensee assuming reverse flow was present could not be supported by the actual plant responses observed. It was therefore concluded that sustained reverse flow was unlikely. These results support the existence of a counter-current type of flow regime upstream of the injection nozzle. Since LOFTRAN does not model this type of mixing the magnitude of flow from the faulted steam generator required to produce the quasi-steady temperature response was estimated from the cold leg inlet temperature and safety injection flow in the faulted loop calculated with LOFTRAN. The results were compared with available thermal-mixing literature. A review of existing experimental data (Creare mixing experiments, References 4 and 5) suggests that, indeed, a significant portion of the safety injection flow into a stagnant loop would propagate upstream of the injection nozzle, and result in the type of counter-current flow observed. The results of this evaluation indicated that a minimum loop flow of 21 lbm/sec existed.

4.5 Reactor Coolant Temperature

Evaluation of the potential flow distributions within the faulted loop cold leg suggested that multi-dimensional behavior may have significantly affected the actual temperature response. However, the LOFTRAN calculations showed faulted loop temperatures significantly below the observed ones. This is believed due to the inability of LOFTRAN to treat complex flow regimes, and the requirement that the pressurizer be in the loop without reverse flow. In addition LOFTRAN was unable to predict the temperature increase observed following isolation of the PORV. Although the calculated faulted loop cold leg inlet temperature was not significantly affected, the LOFTRAN modelling of the pressurizer in the intact loop, may have artificially promoted flow toward the vessel.

In order to estimate the expected minimum temperature in the reactor vessel, the vessel downcomer, cold leg and crossover leg piping, and reactor coolant pump were modelled as a single, mixing volume. The temperature response of this configuration to flow from the faulted steam generator and safety injection flow was calculated assuming perfect fluid mixing. A sustained loop flow rate of 21 lbm/sec was assumed, as indicated above. The calculated minimum temperature was 200°F, as compared with the observed value of 265°F.

4.6 Pressurizer Level

The pressurizer level response obtained from the LOFTRAN calculation compared favorably with the observed data, although some differences were evident. The initial decrease in level was predicted quite well. The data indicated the level returned on span when the charging pumps were started, while LOFTRAN did not predict this to occur until the PORV was first cycled. When the vessel head water began to flash, a rapid filling of the pressurizer was observed and calculated. In the observed data the level went off scale high, while the calculated response indicated the level to still be on scale. This was attributed to an initially lower level calculated by LOFTRAN, and also may have been due to a slightly underestimated voiding of the upper head because of the LOFTRAN modelling which assumes the upper head is a homogeneous region.

4.7 Break Flow

The break flow calculation, primary-to-secondary leakage, used in LOFTRAN assumed an effective break area and a modified Zaloudek critical flow correlation. The LOFTRAN calculated faulted steam generator pressure was underpredicted, as a result of the modelling used. Consequently, secondary-to-primary flow was not calculated by LOFTRAN when the PORV was opened. A more detailed model was developed to assess the limitation of the LOFTRAN break flow model and the effects on the analysis results. It was concluded that the LOFTRAN model, with the exception of reverse flow, provided a reasonable estimate of the break flow. The calculated lower pressure in the faulted loop steam generator results in an overestimate of the primary-to-secondary leakage, which is a conservative result when applied to SAR licensing analyses concerning radiological consequences.

4.8 Reactor Coolant Voiding

The calculated upper head temperature history indicated that voiding may have occurred in the upper head region prior to RCP trip. Although not calculated by LOFTRAN, the void volume was estimated to be less than 132 cubic feet. Any steam bubble in the upper head while the RCPs were running would have been quickly condensed. The observed response indicated this was the case. Upper head voiding was both indicated and calculated when the PORV was opened. The size of the void was estimated to be approximately 305 cubic feet, which is the upper head volume. The observed response indicated at least partial refill of the void, while LOFTRAN effectively inhibits refill. The size of the void when safety injection was terminated could not be determined from available data. Mass balance calculations done by the licensee suggested that a maximum void of 125 cubic feet could have been present at this time.

4.9 Steam Generator Overfill

Primary to secondary leakage in excess of steam flow eventially filled the B steam generator with water and lifted the secondary safety valve. The LOFTRAN calculation indicated overfill of the faulted steam generator and steam line, resulting in lifting of the steam generator safety valve, somewhat earlier than was observed. The early overfill was attributed to the imposition of early termination of steam relief to the condenser in the LOFTRAN calculation in order to better simulate the transient response. This resulted in a calculated steam generator inventory that was about 11,000 lbm higher than actual. In addition, more primary to secondary leakage may have been calculated by LOFTRAN. The combined effect was the earlier filling of the secondary side volumes.

4.10 Long Term Recovery

The LOFTRAN calculation was terminated when safety injection was terminated since the homogeneous equilibrium model on the secondary side overestimated the primary-to-secondary pressure differential and therefore leakage through the failed tube. The remainder of the analysis was based on the actual sequence of events.

5.0 CONCLUSION

The licensee used the LOFTRAN computer program as the primary tool for modelling the steam generator tube rupture incident. While LOFTRAN does have certain limitations (i.e., the inability to calculate complex transients) resulting from its conservative nature as developed for SAR licensing analysis the licensee clearly identified these limitations and performed auxiliary calculations to support and supplement the LOFTRAN results. These auxiliary calculations employed standard mass and energy balance techniques to address the limitations in the LOFTRAN results. These calculations were also reviewed by the staff and found to be acceptable. These analyses support the verification of the system phenomena, including void formation, as required by NUREG-0916.

The staff finds the information provided by the licensee acceptable for the evaluation of the Ginna SGTR event of January 25, 1982.

6.0 ACKNOWLEDGEMENT

E Throm and G. Dick prepared this evaluation.

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REFERENCES

1. Docket No. 50-244, "Response to Safety Evaluation Report-NUREG-0916 Steam Generator Tube Rupture Incident R.E. Ginna Nuclear Power Plant Docket No. 50-244," letter from J.E. Maier to D.M. Crutchfield, November 22, 1982 (DCS 8211290410).

2. NUREG-0916, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant," USNRC, April 1982.

3. L.A. Campbell, et.al., "LOFTRAN Code Description," WCAP-7878 Rev. 3, January 1977.

4. J.A. Block, "Fluid Thermal Mixing in a Model Cold Leg and Downcomer with Loop Flow," CREARE, Inc., Hanover, New Hampshire, EPRI-NP-2312, April 1982.

5. P.H. Rothe, "Transient Cooldown in a Model Cold Leg and Downcomer," CREARE, Inc., Hanover, New Hampshire, EPRI-NP-3118 (Interim Report), May 1983.