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July 14, 1983

Dr. Thomas E. Murley, Regional Administrator
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Subject: Revisions to Emergency Plan Implementing Procedures
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Dr. Murley:

In accordance with 10 CFR 50.54(q), enclosed are two copies of revisions to Ginna Station Emergency Plan implementing procedures, one copy in compliance with the rule and one copy for the Instant Response Center. Two copies are being submitted to the Document Control Desk in Washington.

Very truly yours,

John E. Maier

John E. Maier

Enc. (2)

xc: Document Control Desk (2)

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JUL 22 1983

Docket No.: 50-244

MEMORANDUM FOR: Gus C. Lainas, Assistant Director for Operating Reactors, Division of Licensing

FROM: R. Wayne Houston, Assistant Director for Reactor Safety, Division of Systems Integration

SUBJECT: SAFETY EVALUATION FOR R.E. GINNA NUCLEAR POWER PLANT
(1) Steam Generator Tube Rupture Thermal-Hydraulic Analysis (TACS 49344)
(2) Reactor Coolant Pump Alternate Trip Criteria and Restart Criteria (TACS 49345)
(3) Stagnant Flow Scenarios and Pressurized Thermal Shock (TACS 49351)

Plant Name: R. E. Ginna Nuclear Power Plant, Unit 1
Docket No.: 50-244
TAC Nos.: 49344, 49345 and 49351
Licensing Stage: OR
Requested Date: July 31, 1983
Review Branch: Reactor Systems Branch
Review Status: Completed
Project Manager: G. Dick

The Reactor Systems Branch has completed the evaluation of the subject TACS for the R.E. Ginna Nuclear Power Plant, as provided in the licensee response to NUREG-0916 of November 22, 1982. These Safety Evaluation are provided in the enclosures.

We have found the material provided to be acceptable in addressing the concerns identified in NUREG-0916.

Original Signed By
R. Wayne Houston

R. Wayne Houston, Assistant Director
for Reactor Safety
Division of Systems Integration

Enclosures:
As stated

cc: F. Miraglia
D. Crutchfield
G. Dick

*SEE PREVIOUS CONCURRENCE SHEET

CONTACT: E. Throm, NRR
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EM 7/18/83

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DATE		7/8/83	7/12/83	7/22/83	7/22/83	

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ENCLOSURE 1

STAFF SAFETY EVALUATION CONCERNING ANALYSIS OF PLANT RESPONSE DURING JANUARY 25, 1982 STEAM GENERATOR TUBE FAILURE AT THE R.E. GINNA NUCLEAR POWER PLANT (ITEM 7)

Attachment B of the licensee response (reference 1) to the staff Safety Evaluation Report (NUREG-0916, reference 2) addressed the thermal-hydraulic analysis of the steam generator tube rupture (SGTR) using the LOFTRAN computer program (reference 3). The licensee committed to perform a detailed thermal-hydraulic analysis of system behavior during the incident to verify phenomena, including void formation.

The LOFTRAN computer program has been approved for use in Safety Analysis Reports for analyses of Chapter 15 design basis events (reference 4). 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. These input data and models are not necessarily representative if applied to a real (best estimate) calculation. 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 to provide more detailed modelling of localized events not treated directly in LOFTRAN. The purpose of this evaluation was to verify the thermal-hydraulic phenomena during the incident, including void formation.

LOFTRAN is somewhat limited by the modelling of the upper head region, steam generator secondary side, and primary-to-secondary leakage. The upper head modelling assumes homogeneous, thermodynamic equilibrium conditions during flashing of the upper head fluid. Refilling of the upper head region is artificially constrained to simulate non-equilibrium behavior. Effectively, the upper head region can not refill during natural circulation flow. As a result of this model, LOFTRAN is limited in it's ability to predict void collapse in the upper head. A faster repressurization is calculated which exacerbates the SGTR event by increasing leakage and reducing HPI makeup. Furthermore, flow into the upper head region via guide tubes is not represented. Consequently, the calculated upper head fluid temperature may be unrealistic for plants with small upper head "spray" nozzles, such as Ginna. LOFTRAN is also limited by the homogeneous, saturated conditions within the secondary which promotes an unrealisticly lethargic tube bundle region temperature response to AFW flow and secondary-to-primary heat transfer. In addition, these conditions result in artificially reduced steam generator pressures when no steam flow occurs since the steam is effectively assumed to be in contact with the steam generator tubes. The break flow calculations within LOFTRAN are based on conservative, i.e., maximum flow, critical flow correlations. The accuracy of these correlations in predicting critical flow trends over a wide range of system conditions is uncertain. Furthermore, the break flow modelling does not consider flow resistance through the failed tube, or fluid temperature variations between the steam generator inlet and outlet plenums. Finally, LOFTRAN does not permit reverse flow to occur in the coolant loop to which the pressurizer is connected. For the results presented, the pressurizer was modelled on the intact loop although during the Ginna event the pressurizer was on the faulted loop.

This may result in unrealistic loop flows during refilling of the pressurizer.

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, but was not based on LOFTRAN calculations.

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.

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. Because of the homogeneous equilibrium secondary side model used in LOFTRAN, it was necessary to artificially steam the generator to reproduce the recorded subcooling in the associated cold leg for the transient period from 7 to 16 minutes following the SGTR.

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 overestimated the pressure response by less than 100 psia for the period 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.

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 support 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 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. However, LOFTRAN does not model this type of mixing. A review of existing experimental data (Creare mixing experiments, references 5 and 6) 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.

Reactor Coolant Temperature

The inability of LOFTRAN to treat complex flow regimes, and the requirement that the pressurizer be in the loop without reverse flow, resulted in calculated faulted loop temperatures significantly below the observed values. In addition LOFTRAN was unable to predict the temperature increase observed following isolation of the PORV. 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, a mixing evaluation was performed assuming a sustained loop flow rate of 21 lbm/sec, as indicated above. The calculated minimum temperature was 200°F, as compared with the observed value of 265°F.

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.

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.

Reactor Coolant Voiding

Upper head void, although not calculated by LOFTRAN, prior to RCP trip 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, 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. A mass balance evaluation suggested that a maximum void of 125 cubic feet could have been present at this time.

Steam Generator Overfill

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. This was attributed to early termination of steam relief to the condenser in the LOFTRAN calculation (in order to better simulate the transient response), resulting in about 11,000 lbm more inventory being in the steam generator. 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.

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.

Summary and Conclusions

There are a number of limitations inherent in the LOFTRAN computer program. These limitations, in particular the homogeneous equilibrium model and one dimensional flow models, limit the capability of LOFTRAN to calculate complex transients, such as the Ginna incident. These limitations are the result of the conservative nature of LOFTRAN, as developed for SAR licensing analyses.

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.

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. 50-224," 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. Safety Evaluation Report, "LOFTRAN Code Description," memorandum from R. W. Houston to G. C. Lainas, June 27, 1983.

5. 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.

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

ENCLOSURE 2
STAFF SAFETY EVALUATION CONCERNING
REACTOR COOLANT PUMP ALTERNATE TRIP CRITERIA AND
REACTOR COOLANT PUMP RESTART CRITERIA FOR THE
R.E. GINNA NUCLEAR POWER PLANT (ITEMS 8 & 9)

Attachment C of the licensee response (reference 1) to the staff Safety Evaluation Report (NUREG-0916, reference 2) addressed alternate reactor coolant pump (RCP) trip criteria. Attachment D (reference 1) addressed the RCP restart requirements following a steam generator tube rupture (SGTR).

The alternate RCP trip criteria reviewed included: (1) the current criterion of reactor coolant system (RCS) pressure below 1285 psia (including instrumentation uncertainty), (2) reactor coolant subcooling, (3) a secondary pressure dependent RCS pressure value, (4) reactor vessel level, and (5) reactor coolant pump electrical current.

The reactor vessel level and reactor coolant pump current methods were dismissed because of the need for substantial equipment modification and the need for extensive analytical and experimental efforts.

Several LOFTRAN analyses were performed for a spectrum of SGTR events to assess the margin to RCP trip following a SGTR. It was concluded that the secondary pressure dependent RCS pressure method provided the most potential for preventing RCP trip for a SGTR. It was also concluded that this method was beneficial only if instrument uncertainties were evaluated for normal containment conditions. If abnormal containment condition occur (increased pressure and temperature), then RCP trip could be expected, based on increased instrument uncertainties. Normal containment conditions are expected during a SGTR. It was therefore concluded that "whenever the need for pump trip is addressed in the procedure, the operator would be required to evaluate the containment condition and to select the appropriate criteria depending upon containment conditions. [e.g., for normal containment condition use secondary pressure dependent RCS pressure; for abnormal condition use 1285 psia RCS pressure]. This approach would prevent RCP trip for a design basis SGTR, while still providing for a required pump trip in the event of a LOCA".

In response to Generic Letter 83-10d (reference 3), the licensee (reference 4) has indicated that the resolution for RCP trip will be addressed in a Westinghouse Owners Group (WOG) submittal, scheduled for December 1983. The RCP trip setpoints will be incorporated in Revision 1 of the WOG Emergency Response Guidelines, scheduled for July 31, 1983. The licensee has committed to implement the revised criteria into the existing emergency procedures and provide operator training within 2 months of receipt of the revision (provided the necessary instrumentation is currently available). Based on the information provided in reference 4, the staff believes the necessary instrumentation is currently available in the plant.

Based on the information provided in Attachment C, and on the licensee's commitment to the WOG resolution of the pump trip issue, the staff concludes that the licensee is cognizant of the RCP trip issue and is in conformance with the requirements of Generic Letter 83-10d. The licensee has also evaluated the RCP restart criteria to assess the potential for coolant flashing and loss of pressurizer control during pump restart following a SGTR.

The issue of concern is loss of pressurizer level (and unavailability of pressurizer heaters to control pressure) resulting from the collapse of a sufficiently large steam bubble in the vessel upper head after pump restart. The collapsed bubble draws water from the pressurizer and reduces the reactor coolant subcooling.

The current emergency operating procedures in place at Ginna were reviewed to determine if indicated pressurizer level and reactor coolant subcooling would be maintained follow RCP restart after a SGTR event. It was concluded that the reactor restart criteria are sufficient to ensure both indications are maintained. In some cases, for large enough steam bubbles, the level may decrease below the minimum level required for operation of the heaters. In these cases guidance is provided to restore level using normal charging and safety injection pumps. It was also concluded that the current RCP restart criteria may not be appropriate for other accidents or multiple failure events where safety concerns exist.

Based on the information provided in Attachment D, the staff concludes that the Ginna RCP restart criteria are sufficient within the context of the steam generator tube rupture emergency operating procedures to ensure that indicated pressurizer level and reactor coolant subcooling would be maintained.

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. Generic Letter 83-10d, "Resolution of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps," letter from D. G. Eisenhut to Licensee, February 8, 1983.

4. Docket No. 50-244, "Response to Generic Letter 83-10d, Automatic Trip of Reactor Coolant Pumps R. E. Ginna Nuclear Power Plant Docket No. 50-244," letter from J. E. Maier to D. M. Crutchfield, April 22, 1983.

ENCLOSURE 3
STAFF SAFETY EVALUATION CONCERNING
EVALUATION OF POTENTIAL STAGNATION TRANSIENTS
AND SCENARIOS FOR THE R. E. GINNA NUCLEAR POWER PLANT (ITEM 18 & 19)

Attachment F of the licensee responses (reference 1) to the staff Safety Evaluation Report (NUREG-0916, reference 2) addressed potential situations which could lead to loss of natural circulation following reactor coolant pump (RCP) trip. Loss of natural circulation can result in cold safety injection water entering the reactor vessel downcomer and increase the likelihood of having a pressurized thermal shock (PTS) event.

The situations addressed were: (1) inadequate core heat generation (decay heat fractions less than 0.5 percent of full power), (2) loss of reactor coolant system inventory (small break LOCA), (3) inadequate symmetric heat heat removal (loss of heat sink), and (4) non-symmetric heat removal (overcooling in one steam generator, due to steam line break or steam generator tube rupture resulting in the isolation of one steam generator).

Following a period of loss of natural circulation, starting the RCPs, using the PORV to depressurize the reactor coolant system, or a subsequent break in the hot leg or reactor vessel upper head region can draw the cold safety injection water into the downcomer. During extended periods of loss of natural circulation (on the order of 20 minutes), safety injection will result in low downcomer temperatures.

The staff has evaluated PTS for Westinghouse plants in SECY-82-465 (reference 3) based on the Westinghouse Owners Group PTS program (reference 4). This evaluation included the consequences of the events described above. In SECY-82-465 the staff concluded that the risk associated with PTS, including events which lead to loss of natural circulation, is acceptable for plants with nil-ductility transition reference temperatures below the screening criterion value of 300°F (circumferential welds). The reference temperature calculated for R.E. Ginna was 213°F (as of December 31, 1981).

The Ginna SGTR event of January 25, 1982 was also evaluated in SECY-82-465. It was concluded that for a reference temperature below 378°F, no PTS related vessel failure would occur for this event.

REFERENCES

1. Docket 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. SECY-82-465, "Pressurized Thermal Shock (PTS)," November 23, 1982.

4. "Summary of Evaluations Related to Reactor Vessel Integrity," WOG Letter OG-70, from O.D. Kingsley to H. Denton, May 28, 1982.