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SUBJECT: Responds to 850221 request for addl info to complete safety evaluation re performance testing of safety & relief valves, per NUREG-0737,Item II.D.1.Feedline break analysis performed in 1973 using conservative assumptions.

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May 24, 1985

Director of Nuclear Reactor Regulation
Attention: Mr. John A. Zwolinski, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: NUREG-0737, Item II.D.1 - Performance of Testing of Relief and Safety Valves
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Zwolinski:

Your letter dated February 21, 1985 requested that RG&E provide additional information in order that the NRC Staff can complete a Safety Evaluation of the performance testing of safety and relief valves. A response to each of the Staff requests is contained in Attachment A.

Very truly yours,

Roger W. Kober

Roger W. Kober

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Attachment A

Response to NRC Request for Additional Information
Dated February 21, 1985
NUREG-0737 Item II.D.1

QUESTION 1

The Westinghouse valve inlet fluid conditions report stated that liquid discharge through both the safety and Power Operated Relief Valves (PORVs) is predicted for a FSAR feedline break event. The Westinghouse report gave expected peak pressure and pressurization rates for some plants having a FSAR feedline break analysis. The Robert E. Ginna Unit 1 plant was not included in this list of plants having such a FSAR analysis. Nor does the R. E. Ginna plant specific submittal address the FSAR feedline break event. NUREG-0737, however, requires analysis of accidents and occurrences referenced in Regulatory Guide 1.70, Revision 2, and one of the accidents so required is the feedline break. Provide a discussion on the feedwater line break event either justifying that it does not apply to this plant or identifying the fluid pressure and pressurization rate, fluid temperature, valve flow rate, and time duration for the event. Assure that the fluid conditions were enveloped in the EPRI tests and that the time period of water relief in the EPRI test was as long as expected at the plant. Demonstrate operability of the safety valves and PORV's for this event and assure that the feedline break event was considered in analysis of the piping system.

RESPONSE:

A feedline break analysis was performed in 1973 using very conservative assumptions in order to maximize the time delay to reactor trip and minimize steam generator heat removal capability. Additional conservatism was provided in the analysis by making the assumption that no auxiliary feedwater was available for 10 minutes in order to demonstrate that a manually initiated standby auxiliary feedwater system was sufficient to remove decay heat following postulated accidents. The results were found to be acceptable (see SEP Topic XV-6, NRC SER of 9-04-81).

Although the full diameter feedline break downstream of the feedwater check valve was assumed along with a 10 minute auxiliary feedwater delay for purposes of analysis, such a combination is not required by the Ginna design basis. No full diameter feedwater line break is postulated which can disable all three automatically initiated auxiliary feedwater pumps. Thus this analysis is not appropriate to be used for the purposes of pressurizer relief and safety valve system qualification. Nuclear plants such as Ginna were licensed prior to issuance of Regulatory Guide 1.70, Revision 1, and were not required to consider the feedline break as part of their design basis.

QUESTION 2

In valve operability discussions on cold overpressurization transients, the submittal only identifies conditions for water discharge transients. According to the Westinghouse valve inlet fluid conditions report, however, the PORVs are expected to operate over a range of steam, steam-water, and water conditions because of the potential presence of a steam bubble in the pressurizer. To assure that the PORVs operate for all cold overpressure events, discuss the range of fluid conditions for expected types of fluid discharge and identify the test data that demonstrate operability for these cases. Since no low pressure steam tests were performed for the relief valves, confirm that the high pressure steam tests demonstrate operability for the low pressure steam case for both opening and closing of the relief valves.

RESPONSE:

With regard to the cold overpressure (COP) event, the maximum temperature and pressure conditions that can be achieved at the PORV inlet coincidentally occur for steam bubble operation. Since pressure is normally maintained below the PORV setpoint, the maximum steam and saturated liquid pressure maintained in the pressurizer during startup and shutdown operations in anticipation of the COP event would occur at the PORV setpoint. This pressure is 435 psig, and the corresponding temperature is 456°F. To allow for the calculated potential overshoot of the 435 psig setpoint, the limiting pressure is 535 psig.

Accordingly, the potential worst case scenarios (i.e., maximum temperature and pressure conditions) for PORV discharge during a COP event would be:

1. Discharge of saturated steam at P $\leq$ 535 psig and T $\leq$ 456°F (steam in upper phase of pressurizer).

2. Discharge of saturated water at P $\leqq$ 535 psig and T $\leqq$ 456°F (saturated water in pressurizer).

3. Discharge of subcooled water at P $\leqq$ 535 and T $\leqq$ 456°F (some mixing of colder RCS water with saturated pressurizer water).

4. Scenario 1 followed by Scenario 2

5. Scenario 2 followed by Scenario 3

6. Scenario 1 followed by Scenario 2 followed by Scenario 3.

EPRI test conditions for the PORV's were chosen based on expected inlet fluid conditions. Tests were limited but designed to confirm operability over a full range of expected inlet conditions. Steam, steam-to-water and water flow tests were conducted. Results of these tests can be found in EPRI report EPRI NP-2670-LD, Volume 8. Although steam tests were conducted only at the higher pressures, that satisfactory operation would also result at the less severe lower pressures. This can be confirmed by the high pressure versus low pressure water tests where successful valve operation was observed.

QUESTION 3

Results from the EPRI tests on the Crosby safety valve indicate that the test blowdowns exceeded the design value of 5 percent for both "as installed" and "lowered" ring settings. If the blowdowns expected for the plant (see Question 4) also exceeded 5 percent, the higher blowdowns could cause a rise in pressurizer water level such that water may reach the safety valve inlet line and result in a steam-water flow situation. Also the pressure might be sufficiently decreased such that adequate cooling might not be achieved for decay heat removal. Discuss these consequences of higher blowdowns if increased blowdowns are expected.

RESPONSE:

The impact on plant safety of pressurizer relief valves blowdowns in excess of 5 percent for R. E. Ginna was evaluated. The results of this evaluation showed no adverse effects on plant safety.

Relief valve blowdowns in excess of that assumed in the R. E. Ginna Reload Transition Safety Report (RTSR) (submitted by letter dated December 20, 1983 from John E. Maier, RG&E, to Harold R. Denton USNRC) will have the following effects on the events in which relief valve actuation occurs:

1) Increased pressurizer water level during and following the valve blowdown,
2) Lower pressurizer pressure during and following valve blowdown,
3) Increased inventory loss through the relief valve.

The impact of the increased relief valve blowdowns with respect to the above effects was evaluated for the two R. E. Ginna RTSR events in which relief valve actuation occurs, (i.e., Loss of Load and Locked Rotor).

For the Loss of Load event, results from sensitivity analyses performed by Westinghouse for 4 loop plants were used for the evaluation. Similar results are expected for 2 loop plants. These analyses investigated the effects of different blowdown rates on the Loss of Load event. The results of these analyses showed only marginal increases in pressurizer water volume and the maximum pressurizer water levels were well below the level at which liquid relief would occur. Peak RCS pressures were shown to be unaffected by the increased blowdowns. The increased blowdowns did result in lower pressurizer pressure and increased RCS inventory loss, however, these had no adverse impact on the event and adequate decay heat removal was maintained.

For the Locked Rotor event, increased relief valve blowdowns have little impact on the event. As analyzed and presented in the R.E. Ginna RTSR, the opening and closing of the relief valve occurs over a short time period (3 seconds). As a result, there is little change in either pressurizer level or RCS inventory. Increased relief valve blowdowns would have no impact on peak pressure, peak clad temperature, or DNBR as these occur prior to the closing of the relief valve.

QUESTION 4

The submittal states that an evaluation of safety valve ring settings is being performed by RGSE and Westinghouse to verify that the ring settings used will result in proper valve performance on a plant specific basis. Identify the final ring settings selected as a result of this evaluation. Since EPRI tests on the Crosby 3K6 and 6M6 safety valves were used to assess performance of the 4K26 valve of R. E. Ginna, identify which EPRI tests on the 3K6 and 6M6 valves had ring settings representative of those used on the plant 4K26 valve. Identify the expected blowdowns corresponding to the plant ring settings and explain how these blowdowns were extrapolated or calculated from test data. Verify that with the ring settings used the valves can perform their pressure relief function and the plant can be safely shutdown with the blowdown, back pressures, and fluid conditions occurring at the plant.

RESPONSE:

Valve ring settings for the Ginna safety valves were developed using Crosby production test methods and will have performance characteristics similar to those valves tested at EPRI with "as-shipped" ring settings. Details of these tests were discussed in the original submittal and can be found in EPRI reports EPRI NP-2770-LD, Volumes 5 and 6. Blowdowns measured on the Ginna valves at Crosby during the production testing for each valve were equal to or less than 5 percent.

QUESTION 5

Results from EPRI tests on the Crosby 3K6 and 6M6 safety valves were used to evaluate performance of the Crosby 4K26 valve of R. E. Ginna. The EPRI test results indicate that steam flow rates in excess of rated flow were achieved. A flow rate determination for the R. E. Ginna valves, however, depends on the specific ring settings used at the plant. Thus, provide a demonstration that the plant safety valves will pass their rated flow at the ring settings used.

RESPONSE:

As noted in Table 4.4 of EPRI Report NP-2770-LD, Volume 6, the Crosby 6M6 test valve achieved rated flow for each of the tests reported at 3 percent accumulation regardless of the ring setting used in the test. A review of EPRI Tables 4-3 and 4-4 in Volume 5 of EPRI Report NP-2770-LD reveals that for steam tests of the 3K6 valve where blowdown was measured to be less than 10 percent, flow rates of 119-122 percent of rated flow at 3 percent accumulation were reported. The EPRI tables indicate that lower than rated flows occurred at blowdowns greater than 15 percent. Crosby standard production tests indicate 5 percent blowdown with the "as-shipped" ring settings. These are the ring settings currently used on the installed valves at Ginna Station. This is within the range of both the 3K6 and 6M6 tests where rated flow was achieved; therefore, rated flow can be expected for the 4K26 R. E. Ginna valves.

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QUESTION 6

During an EPRI loop seal steam-to water transition test on the 3K6 valve the valve fluttered and chattered when the transition to water occurred. The test was terminated after the valve was manually opened to stop chattering. The 6M6 valve exhibited similar behavior on two hot loop seal-steam tests and one subcooled water test. Again, these tests were terminated after the valve was manually opened to stop chatter. The hot loop seal tests appear to be representative of conditions at the Ginna plant and the liquid flow tests may be representative of a feedline break event (see Question 1). Just that the valve behavior exhibited in these tests is not indicative of the performance expected for the R. E. Ginna valves.

Response:

The 3K6 steam-to-water transition test was conducted using ring setting that resulted in blowdowns in excess of 15 percent and is, therefore, not considered representative of the ring settings "as-shipped" for the R. E. Ginna valves for which production tests have shown resulted in 5 percent blowdown and stable valve performance.

The two 6M6 loop seal steam tests 920 and 1419 which ended in valve chatter were repeats of two other successful steam tests, 917 and 1415. Close examination of these tests reveals that the

valve opened on demand, relieved the test system pressure transient for approximately 20 seconds, reduced the initial pressure by 4 to 5 percent as required and then closed. Upon closing the valve-reopened only on tests 920 and 1419 and chattered. Valve inlet, pressure plots show that pressure oscillations at the valve inlet, which vary in magnitude from the set pressure to the blowdown pressure, drove the valve to chatter. These pressure oscillations are acoustic in nature and are due to the long loop seal length tested by EPRI. Since the R. E. Ginna inlet piping is shorter than the EPRI piping, more stable results would be expected at Ginna.

The Crosby 6M6 test valve experienced chattering during the 500°F subcooled water test. This is not unexpected as spring loaded safety valves such as the Crosby 6M6 and 4K26 are designed to operate on steam and rely on the fluid expansion for proper operation. Subcooled water relief is provided by the R. E. Ginna PORV's.

QUESTION 7

NUREG-0737, Item II.D.1 requires that the plant-specific PORV control circuitry be qualified for design-basis transients and accidents. Please provide information which demonstrates that this requirement has been fulfilled.

RESPONSE:

NUREG-0737 Item II.D.1 required that licensees conduct testing to quality the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. Licensees were to determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences. The single failures applied to these analyses were to be chosen so that the dynamic forces on the safety and relief valves were maximized. Test pressures were to be the highest predicted by conventional safety analysis procedures. The required analyses were performed and the expected valve operating conditions were given in the RGE report submitted March 4, 1983. The limiting conditions were developed based upon FSAR, extended high pressure injection and cold overpressurization events. The limiting events for two loop plants, including Ginna, were loss of load and locked rotor transients.

No relief or safety valve discharge will occur for extended high pressure injection events. Cold overpressurization events will not cause adverse environmental conditions in containment. The limiting pressure event, a locked rotor, is analyzed in Section 15.3 of the Ginna UFSAR. The transient is terminated by safety valve operation discharging at 20 ft^3/sec over a period of approximately four seconds. The discharge of the safety and relief valves is to the pressurizer relief tank which is designed to condense and cool a discharge equivalent to 110 percent of the full power pressurizer steam space (UFSAR Section 5.4.8) or approximately 400 ft^3. Thus, no adverse environment is expected to occur as a result of the most limiting transient requiring operation of the relief or safety valves. Those transients that do cause an adverse environment, such as LOCAs, steam line breaks, feedwater line breaks or other high energy line breaks do not require operation of the PORVs to terminate the transient or mitigate the consequences of the event. Therefore, the Ginna power operated relief valves (PORVs) are capable of fulfilling their design function.

In response to 10 CFR 50.49, RG&E provided, in a submittal dated August 30, 1984, a discussion of the selection process used to define electrical equipment required to operate in a harsh environment. Based upon that methodology, which was found acceptable in the NRC's SER dated February 28, 1985, RG&E has determined that the pressurizer PORVs and control circuitry are not required to be qualified to operate in a harsh environment.

QUESTION 8

Bending moments are induced on the safety valves and PORVs during the time they are required to operate bacause of discharge loads and thermal expansion of the pressurizer tank and inlet piping. Make a comparison between the predicted plant moments with the moments applied to the tested valves to demonstrate that the operability of the valves will not be impaired.

RESPONSE:

The bending moments induced on the safety and relief valves tested by EPRI exceeded the bending moments predicted for the Ginna safety and relief valves. The maximum moment tested for the 6M6 valve was during test 908 and was 298.75 in-K. The largest moment predicted for the safety valves for Ginna is 73.27 in-K, thus demonstrating functionability for the Ginna safety valves.

Likewise, a bending moment of 43.0 in-K was induced at the inlet of the Copes-Vulcan PORV test valve per EPRI test 64-CV-174-2S. The largest bending moment predicted for the Ginna relief valves is 30.48 in-K, thus demonstrating functionability for the Ginna relief valves.

QUESTION 9

The submittal does not address the NUREG-0737 II.D.1 requirement that the operability of the PORV block valves be demonstrated. A test program on block valves was performed, though, which is described in the EPRI/Marshall Electric Motor Operated Valve Interim Test Data Report. In this test program the Limitorque SMB-000-5 motor operator that is used at the R. E. Ginna plant was not tested. Since the SMB-000-5 operator is smaller than any tested, explain how these tests results or other test data can be used to demonstrate operability of the motor operator.

RESPONSE:

Three differences exist between the R. E. Ginna block valves and those (Velan) block valves tested by EPRI. First the motor operator installed on the R. E. Ginna block valves is a Limitorque SMB-000-5 versus the Limitorque SB-00-15 operator used with the EPRI test valve. The speed of the two valves is different, 40 seconds for the R. E. Ginna valve and 10 seconds for the EPRI valve and finally the seat bore of the EPRI valve (2.625 inches) is larger than that of the R. E. Ginna (2.25 inches) valve. The R. E. Ginna block valve operators provide an output thrust at the as-shipped torque switch setting that exceeds that required to close by approximately 26 percent.

Since the valves at R. E. Ginna are of similar design except for the motor operators (the motor operators are different due to the different closing time requirements) to the ones tested by EPRI and the output thrust is greater than required, the valves at R. E. Ginna will exhibit performance equal to or better than the EPRI test valves. The fact that the seat bore is smaller on the Ginna block valve results in a smaller thrust required to close the valve as compared to the test valve at the same differential pressure.

QUESTION 10

The submittal indicates that a discharge piping backpressure of 350 psig is developed. Describe how this value for backpressure was determined. Among the EPRI tests performed on the Crosby 3K6 and 6M6 safety valves, thre was only one test performed with a hot loop seal and high backpressure. This was a loop seal-steam-to-water transition test on the 3K6 valve, which resulted in valve chatter. Explain how the results of cool loop seal tests with high backpressure can be used to show that the safety valves of the Ginna plant can successfully discharge hot loop seal water followed by steam under high plant backpressures.

RESPONSE:

The maximum safety valve discharge piping backpressure during and following discharge of the safety valves calculated from our thermal hydraulic analysis is less than 357 psia. The thermal hydraulic analysis was described in the submittal dated March 4, 1983 from John E. Maier, RG&E to Dennis M. Crutchfield, USNRC.

The EPRI tests showed that the Crosby safety valves always lifted on steam after loop seal water discharge whether it be hot or cold. Dynamic backpressures measured during the EPRI tests were approximately 9 to 29 percent of the setpoint; however, Crosby valves employ a balanced bellows design that is relatively insensitive to backpressure effects as noted by EPRI on page S-5 of EPRI report NP-2770-LD, Volume 5. Backpressure effects and loop seal water temperature, therefore, should not impact valve functionability.

QUESTION 11

In the analysis of the hot loop seal discharge case, which is representative of the R.E. Ginna installation, the loop seal temperature distribution was assumed to be consistent with the distribution of EPRI test 917. A misrepresentation of the temperatures in the loop seal could have a significant effect on the calculated forces acting on the piping system. Therefore, provide verification that the temperatures assumed for the loop seal are accurate.

RESPONSE

RG&E does not currently have data verifying that the temperatures in the loop seal are consistent with those used in the safety valve discharge piping analysis. Therefore, in order to provide assurance that the temperatures in the loop seal are representative of those assumed in the piping analysis, further work has been initiated. During the 1985 refueling outage at Ginna physical measurements in the loop seal area were taken to permit the performance of a detailed thermal analysis and design of an improved insulation system for the safety valve loop seals. This new insulation system will ensure that the temperatures achieved are consistent with those assumed in the previous piping analysis. Installation of this modification is presently scheduled for the 1986 refueling outage.

QUESTION 12

According to results of EPRI tests, high frequency pressure oscillations of 170-260 Hz typically occur in the piping upstream of the safety valve while loop seal water passes through the valve. The submittal refers to an evaluation of this phenomenon that is documented in the Westinghouse report WCAP 10105 and states that the acoustic pressures occurring prior to and during safety valve discharge are below the maximum permissible pressure. The study discussed in the Westinghouse report determined the maximum permissible pressure for the inlet piping and established the maximum allowable bending moment for Level C Service Conditions in the inlet piping based on the maximum transient pressure measured or calculated. While the internal pressures are lower than the maximum permissible pressure, the pressure oscillations could potentially excite high frequency vibration modes in the piping, creating bending moments in the inlet piping that should be combined with moments from other appropriate mechanical loads. Provide one of the following: (a) a comparison of the allowable bending moments established in WCAP 10105 for Level C Service Conditions with the bending moments induced in the plant piping by dynamic motion and other mechanical loads or (b) justification for other alternate allowable bending moments with a similar comparison with moments induced in the plant piping.

RESPONSE

The R.E. Ginna piping system response including the safety valve loop seal region is due to frequencies less than 100 HZ. The

frequency of the forces and moments in the 170-260 HZ range potentially induced by the pressure oscillations is significantly greater than this frequency. The upper limit of significant content for similar systems is also much less than this (170-260 HZ) range. Industry data indicates that only frequencies of 100 HZ or less are meaningful. The EPRI data confirms this. Consequently, no significant bending moment during the pressure oscillation phase of the transient will occur.

In the submittal, pressure stresses based upon a design pressure of 2580 psig were included with the bending moments resulting from the deadweight and the safety valve discharge piping loads. Because of the time phasing of the pressure oscillation (during water slug discharge through the safety valve) and the discharge piping loads (subsequent to water slug discharge through the valve) this pressure term and moment term were not added. They do not occur coincidentally. A comparison of the intensified bending moments from the stress evaluation and the allowable moment presented in WCAP-10105 shows that all values are below the allowables. Specifically, the maximum allowable moment from Table 4-7 of WCAP-10105 for 4-inch schedule 160 piping for an internal pressure of 5000 psi is 176 in-kips. The moments for the sum of deadweight and water slug discharge for the components listed in Table 6-14 of the submittal at nodes 6110, 5010, and 6120 respectively, are 47.2, 52.3, and 47.3 in-kips.