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JOHNSTON,W.V. Region 1, Ofc of the Director

SUBJECT: Responds to NRC 880610 ltr re violations noted in Insp Rept 50-244/88-10.Corrective actions:

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June 29, 1988

William V. Johnston, Acting Director
Division of Reactor Safety
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Subject: Response to Inspection Report 88-10
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Johnston:

The purpose of this letter is to reply to several issues noted in the NRC's June 10, 1988 Inspection Report 88-10, which were also discussed during the June 24, 1988 Enforcement Conference at Region I headquarters. These issues concern the safety significance of omissions of the main feedwater check valves and turbine driven AFW steam admission valves from the Ginna IST program, and RG&E opinion that the steam admission check valves would have operated if called upon, even though the check valve 3504B disc was found not completely free to move.

As a result of the attached evaluations, RG&E believes that, even though the valves had not been included in the IST program, there is reasonable assurance that the valves would have been able to perform their safety function. Furthermore, in the case of the turbine driven AFW steam admission valves, the postulated failure of the valves is bounded by the Ginna Chapter 15 UFSAR Safety Analysis. Thus, although we recognize the significance of not having included the subject valves specifically in the Ginna IST program, we believe that the safety significance relative to the plant accident analysis is minor.

IE01

We trust that this letter is responsive to the requests for information made in the June 10 Inspection Report and at the Enforcement Conference, and can be used in your consideration of enforcement actions.

Very truly yours,

Bruce A. Snow
Superintendent
Nuclear Production

Attachment

xc: U.S. Nuclear Regulatory Commission (original)
Document Control Desk
Washington, DC 20555

xc: Ginna Senior Resident Inspector

ATTACHMENT 1

SAFETY SIGNIFICANCE EVALUATION FOR TDAFW STEAM ADMISSION VALVES 3504B, 3505B

RG&E has evaluated all Chapter 15 accident analyses, wherein the turbine-driven auxiliary feedwater system could be called upon to operate. The postulated accidents, and the effects of the steam admission check valves failing to close, are described below. Based on these evaluations, RG&E has concluded that the steam admission check valves' failure is bounded by the present Chapter 15 accident analyses.

A. SPECTRUM OF STEAM LINE BREAKS

In terms of containment and DNBR effects, the Ginna UFSAR, Section 15.1.5, has determined that the limiting break is inside containment, at the discharge of the steam generators. The reason for this is that the steam lines have nozzles near the outlet of the steam generators, whose functions during normal operation is to measure steam flow. During accident conditions (postulated steam line breaks), the nozzles serve as flow restrictors, limiting flow area to a maximum of ~1.4 ft^2 (vs. 4.37 ft^2 at the outlet). This limits cooldown of RCS, and mass and energy blowdown into containment.

1. Core Return to Power (DNBR Limits)

The Chapter 15 UFSAR analysis (Section 15.1.5.2) is performed for a break size of 4.6 ft^2, compared to the actual size of 4.37 ft^2. An additional 6" line blowdown (0.19 ft^2) would result in a 4.56 ft^2 diameter break, which is bounded by the analyzed break. Thus, there is no effect on the UFSAR analysis.

RG&E has also performed an analysis to determine the effect on DNBR due to blowdown of the isolated steam generator. For this purpose, RG&E analyzed a 4.37 ft^2 steam line break at hot zero power, and compared it with the same case, with additional blowdown through the 6" steam admission lines.

The most limiting DNBR will occur around the time of peak power. At the time of peak power the important DNBR parameters and their effect on DNBR is listed below.

Core power; slightly greater for the added 6" line flow case resulting in slight decrease in DNBR

Core flow; slightly greater for the added 6" line flow case resulting in slight increase in DNBR

Core pressure; slightly less for the added 6" line flow case resulting in slight decrease in DNBR

Core inlet temperature; slightly less for the added 6" line flow case resulting in slight increase in DNBR

It is apparent that there are competing effects on DNBR, due to variations in these parameters. Using the DNBR sensitivities for these parameters associated with the Improved Thermal Design Procedure for Ginna to convert power, flow, pressure, and temperature into DNBR effects results in a negligible effect on DNBR, and remains well above the design limit of 1.3.

Thus, because the effects of this relatively small additional blowdown are negligible, and the limiting analyzed Chapter 15 break size bounds the postulated additional failure, RG&E has concluded that there is no safety significance for this case, relative to the Ginna accident analysis.

2. Containment Pressure Analysis

The peak containment pressure case in the UFSAR is a 4.37 ft^2 break at hot 0% power, with no loss of offsite power. This latter assumption increases the energy release to containment, due to forced reactor coolant system flow. Hot zero power results in a large increase in secondary side inventory, which would also cause greater mass and energy release to containment. Under these circumstances, the TDAFW system would not be actuated, since it requires low-low level in both steam generators, or a loss of offsite power, to actuate the system.

The limiting containment analysis, which could actuate the TDAFW system, is a hot 0% power break, with maximum steam generator inventory. RG&E analyzed containment conditions for this case, with additional blowdown through the 6" TDAFW cross-connect line. The peak containment pressure was determined to be well below the limiting UFSAR case break, and the 60 psig containment design pressure.

Thus, the analysis is bounded by the worst-case break in the Ginna UFSAR.

B. FEEDWATER LINE BREAKS

Feedwater system pipe breaks are discussed in Section 15.2.7 of the UFSAR. This event is analyzed to determine the following:

1. Primary system temperature and pressure response, and
2. Capability of system to remove core decay heat.

The "base" case feed line break results in:

1. Blowdown of steam and liquid from the failed steam generator, initially removing heat from the RCS.

2. Reactor trip on low-low level on the failed steam generator, with level in the intact steam generator conservatively assumed to be at the low level steam flow/feed flow mismatch setpoint. This minimizes steam generator inventory for decay heat removal.

3. The intact steam generator removes decay heat through the relief valves until inventory is lost.

4. The difference in decay heat produced, and the energy removed through the steam generator relief valves is removed from the RCS through the pressurizer relief and safety valves. This continues until such time as auxiliary feed flow is sufficient to remove all decay heat produced (27 minutes).

The purpose of the feed line break analysis is to determine maximum heatup rates of the RCS, and the capability of the pressurizer safety valves to limit the reactor coolant system overpressure. The postulated failure of the turbine-driven AFW steam admission valves would provide another flow path for steam relief, both from the intact and through the affected steam generator. This effect increases heat removal from the reactor coolant system, and thus would limit RCS overpressurization, minimizing the effects of the feedwater line break accident.

Thus, RG&E has concluded that postulated failure of the TDAFW steam admission check valves during a feedwater line break accident is bounded by the present Ginna UFSAR Chapter 15 accident analysis.

C. STEAM GENERATOR TUBE RUPTURE (SGTR)

In the Ginna UFSAR analysis of the SGTR (Section 15.6), it is assumed that the affected generator is detected and isolated in 30 minutes. The turbine-driven AFW system would be expected to be actuated during this event. Prior to the time the affected steam generator is isolated, the Ginna UFSAR analysis assumes reactor coolant release from the secondary

relief valves and/or safety valves. There is no specific location for the release points. If it is postulated that check valves 3504B or 3505B were to fail in the open position, it could be conservatively postulated that flow between the affected and unaffected steam generators could occur, resulting in release of steam from the unaffected steam generator.

This is not considered to extend the bounds of the UFSAR analysis, however, because the amount of reactor coolant released would not change. The Section 15.6 offsite dose analysis makes no distinction relative to the specific location of reactor coolant release. Prior to the required 30 minutes, the unaffected steam line would be isolated from the affected SG by the motor operated valves in the cross-connect (Emergency Procedure E-3, Step 3). This would terminate flow to the unaffected steam generator.

RG&E has recently submitted a revised steam generator tube rupture analysis to the NRC utilizing new Westinghouse Owners Group methodology incorporating operator actions. In that analysis, reactor coolant from the rupture is assumed to be released to atmosphere, until the affected steam generator is isolated. This would be accomplished by use of the motor operated valves in the TDAFW admission cross-connect, similar to the above case.

Thus, for the SGTR, using either methodology, the failure to include the 3504B and 3505B check valves into the IST program would not have any safety consequences relative to the Ginna Chapter 15 FSAR analysis.

ATTACHMENT 2

FEEDWATER CHECK VALVES 3992 AND 3993

As noted during Inspection 88-10, these check valves are not in the Ginna IST Program. Although not explicitly inspected or tested to determine the capability to ensure auxiliary feedwater isolation from the feedwater system, Ginna's normal startup practices provide substantial assurance that this function could have been met. This assurance is determined in the following manner:

During startup operations, steam generator level (below ~3% power) is maintained with the auxiliary feedwater system. The condensate and feedwater system is lined up for power operation, and all manual isolation valves are opened in order to provide a flowpath for feedwater chemistry cleanup. During this phase of startup, valves 3992 and 3993 provide the only isolation barrier between the steam generators and the feedwater cleanup system. Significant back-leakage through the check valves would be noticed by an obvious reduction in normal steam generator level. This has not been Ginna's experience.

Although RG&E cannot quantify the small amounts of leakage that could occur past the check valves under these circumstances, we are confident that significant leakage (>30 gpm) would not have occurred. Since the AFW system is controlled at 230 gpm, and the accident analysis assumes a flow of 200 gpm, 30 gpm margin exists.

Conclusion:

Although these check valves were not in the Ginna IST program, RG&E has confidence that, as a result of normal operating procedures noted above, and the valve disassembly which occurred in 1986 and 1987 and demonstrated unobstructed stroking capability, that the check valves could have performed their auxiliary feedwater diversion isolation function. This annual procedural method is consistent with the IST program frequency, since the valves cannot be stroked during power operation. Thus, although RG&E admits to not having these valves in the program, reasonable assurance of their operability has existed, on an annual basis, and the safety significance of this omission is considered low.

ATTACHMENT 3

CHECK VALVE 3504B CLOSURE FORCE

AFW turbine steam admission check valve 3504B was reviewed for the ability of reverse steam flow to close a stuck open valve. This 6" swing-type check valve contains an internal body stop which limits disc travel in the full-open position. Assuming a disc is stuck open in the full-open position, it will still protrude into the flow stream and will be subjected to pressure forces generated by the flow reversal. In a preliminary calculation, this initial closing force has been determined to be approximately 250 lbs. (applied to the protruding edge of the full-open disc). As the disc is pulled further into the flow stream, the closing force will increase as a larger portion of the disc is acted on by the reverse flow. This 250 lb. initial closing force would exert a torque of approximately 80 ft. lbs. on the hinge pin/actuator shaft and would be comparable to a manually-exerted force of more than 80 lbs. applied at the external counterweight. This is substantially more than was reported to have been required by plant personnel to close the stuck open valve.