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Constellation Energy
Generation Group

May 28, 2005

Ms. Donna M. Skay
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Application of 10 CFR 50.90 Process for Use of Fracture Mechanics Analysis per GDC-4
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

References: (a) Letter from Joseph A. Widay, Ginna LLC to Mr. Robert L. Clark, USNRC, "Fracture Mechanics Analysis per GDC-4", September 30, 2004

Dear Ms. Skay:

The information in Attachment 1 is provided to respond to the April 29, 2005 request for additional information (RAI) from the NRC staff regarding Reference (a).

The information in Attachment 2 augments Reference (a) with an application for use of the 10 CFR 50.90 process of the NRC regulations. The submittal requests that the NRC approve the use of "leak-before-break" for the accumulator lines and pressurizer surge line, as demonstrated by the analyses enclosed in Reference (a) and the response to RAIs provided in Attachment 1 of this letter.

It has been determined that this amendment application does not involve a significant hazard consideration as determined by 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

Approval of this amendment application is requested by September 30, 2005 to provide adequate time to incorporate this change into Revision 19 of the Ginna Station UFSAR.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated New York State Official.

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If you have any questions regarding this submittal, please contact George Wrobel, Nuclear Safety and Licensing at (585) 771-3535.

Very truly yours,

Thomas A. Marlow

Thomas A. Marlow

STATE OF NEW YORK :
: TO WIT:
COUNTY OF WAYNE :

I, Thomas A. Marlow, being duly sworn, state that I am Plant General Manager – R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this response on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Thomas A. Marlow

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Monroe, this 28 day of May, 2005.

WITNESS my Hand and Notarial Seal: Michalene A. Bunts

Notary Public

MICHALENE A BUNTS
Notary Public, State of New York
Registration No. 01BU6018576
Monroe County
Commission Expires Jan 11, 2007

My Commission Expires:

1-11-07

Date

Attachments

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ATTACHMENT 1

REQUEST FOR ADDITIONAL INFORMATION

R. E. GINNA NUCLEAR POWER PLANT

DYNAMIC EFFECTS OF POSTULATED HIGH-ENERGY LINE BREAKS

By letter dated September 30, 2004, R. E. Ginna Nuclear Power Plant, LLC submitted for Nuclear Regulatory Commission (NRC) staff review and approval analyses regarding the dynamic effects of postulated high-energy line breaks for the pressurizer surge line and accumulator A and B lines at the R. E. Ginna Nuclear Power Plant. These analyses are documented in two reports: WCAP-16311-P, Revision 0, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the R. E. Ginna Nuclear Power Plant," and SIR-99-036, "Leak-Before-Break Evaluation of Portions of the Accumulator A and B Piping at R. E. Ginna Nuclear Power Station," June 1999. To complete its review, the staff requests additional information regarding both reports.

Questions regarding analysis of the Pressurizer Surge Line

The following questions refer to sections of WCAP-16311-P.

1. Section 2.1, states that Alloy 82/182 weld material is not found in the pressurizer surge line; however, the report did not discuss the weld materials. Discuss the materials used in all welds in the surge line and indicate whether safe ends or thermal sleeves are used at these joints.

Response: The weld material type is stainless steel with Gas Tungsten Arc Weld and Shielded Metal Arc Welding processes. There is a stainless steel safe end piece between the nozzle and reducer at the pressurizer nozzle location. There is also a thermal sleeve in the nozzle location.

2. Section 2.4 states that there has been no service cracking or wall thinning identified in the pressurizer surge line of Westinghouse pressurized-water reactors. Discuss the results of past inservice inspection(s) on the Ginna surge line.

Response: As part of the Ginna Inservice Inspection Program, welds in the pressurizer surge lines are periodically inspected. During 2003 refueling outage (RFO), inservice inspection was performed on the stainless steel weld that connects the safe end (Type 316 SS) to the pressurizer surge nozzle (SA 216 WCC). Liquid penetrant test (PT) indications were found on the outer diameter surface of the stainless steel weld near the nozzle/weld metal boundary. A boat sample was removed from a section containing a number of indications and sent to BWXT Services, Inc. (Lynchburg, VA) for laboratory examination to determine the cause of the indications. Findings revealed with reasonable assurance that the weld cracks were the result of hot cracking which developed during original construction. Similar indications were observed during the 2005 RFO inspection of the same weld and these were ground out until the linear indications disappeared.

3. The material properties for the surge piping are listed in Tables 3-1 and 3-2. Confirm that limiting material properties were used when performing the limit load analysis.

Response: Tables 3-1 and 3-2 contain the material properties for the SA376 TP 316 base material. The limiting material properties (minimum yield and ultimate strengths) from the CMTRs (Certified Material Test reports) were used for the limit load analysis.

4. Page 4-4 states that the second highest stress weld is located at Node 1120, which is located on a straight section of the pipe as shown in Figure 4-1 of the report. Discuss why a straight section of pipe has incurred high stresses since high stresses usually are located at nozzles, tees, or elbows, not at a straight section of the pipe.

Response: The highest stress occurs at Node 1020 which is located at the end of the nozzle. The second highest stress location is at Node 1120 which is located at a weld location and it is at the end of the 5-D bend of the pipe. There are no elbows in the surge line.

5. Table 4-2 provides the normal and faulted loading cases for leak-before-break evaluations. Discuss whether the pressurizer reflood transient is included in the limit load analysis.

Response: Pressurizer reflood transient is the term used for the CE plants. However, similar transient loading is included in the limit load analysis.

6. Section 6.2 discusses the results of the fatigue crack growth analyses. The report states that the initial flaw sizes were assumed to be 10% of the wall thickness. However, in leak-before-break analyses for other nuclear plants, Westinghouse assumed several initial flaw sizes. With this limited assumption (i.e., small and only one flaw size), explain why the fatigue growth assessment is adequate to assess fatigue cracking and associated growth. Discuss why several initial flaw sizes were not assumed for Ginna.

Response: Westinghouse methodology for the pressurizer surge line LBB analyses for other plants is to use a single initial flaw size (10% of the wall thickness) to assess the potential fatigue crack growth for a representative location and at various angular locations along the same cross-section.

Table 6-1 shows the results of the fatigue crack growth (FCG) and the FCG is insignificant for the life of the plant. We believe that the FCG assessment performed and shown in the WCAP report is representative and adequate.

7. Section 6.2 states that the 60-year design transients and cycles are the same as those of 40 years for the Ginna surge line. Discuss the basis for this statement.

Response: Through an exhaustive evaluation of plant records (operation logs, computer data, LERs, non-conformance reports, etc.) and interviews with past operating personnel, Ginna compiled a reasonably accurate list of plant transients that occurred since Ginna went critical in

1969 and up to the recent RFO (2005). Conservative extrapolation of these data into the extended period of operation (60 years) showed that these are less than the original design transients and cycles (40 years) that are listed in the UFSAR.

8. Section 7 states that the leak detection capability is 0.25 gpm per hour at Ginna. In recent industry experience, improved fuel integrity and reduced reactor coolant system (RCS) radioactivity levels have caused the gaseous channel of the containment atmosphere radiation monitor to become less effective for RCS leakage detection. It could take longer to detect RCS leakage than is required in the plant technical specifications. For example, the time required to detect a 1 gpm leak with the containment atmosphere gaseous radiation monitors ranges from 223 to 839 hours at a domestic nuclear plant, while the time required to detect the same leak with the particulate monitors now ranges from 3.6 to 7.3 hours. In light of recent industry experience, discuss how the current leak detection system is capable of detecting 0.25 gpm per hour in the containment, and demonstrate how the redundancy, reliability, and sensitivity criteria as recommended in Standard Review Plan 3.6.3.III.3 and Regulatory Guide 1.45 are satisfied.

Response: Ginna is aware of recent experiences where RCS leak detection systems have not been shown to meet the standards of Regulatory Guide 1.45. However, Ginna has demonstrated conformance with that guidance. In our response of August 8, 1998 to RAIs related to our application for leak-before-break to the RHR lines, we explained that our particulate monitors were demonstrated to be capable of detecting very small RCS leak rates, even with robust fuel (we do not credit the gaseous monitors to meet the 0.25 gpm detection capability, although they are a useful backup). The second credited leak detection system is Inventory Balance. The NRC's SER of February 25, 1999 acknowledged these capabilities, which were attributed to the relatively small containment volume, effective recirculation of air in containment, and the second generation of R-11 detector.

Questions regarding analysis of the Accumulator Lines

The following questions relate to SIR-99-036, "Leak-Before-Break Evaluation of Portions of the Accumulator A and B Piping at R. E. Ginna Nuclear Power Station."

9. Section 1.1 discusses the portions of the accumulator piping included in the leak-before-break (LBB) analyses:

   (a) For Accumulator B piping, only nodes 60 and 80 were included which denote both ends of an elbow. Discuss why the rest of the Accumulator B line was not included in the analysis.

   (b) For Accumulator A piping, only node 856 was included. Confirm that the LBB analysis considered only the welds in both Accumulator A and B piping and that no pipe segments were considered.

(c) Confirm that the welds, instead of pipe segments, were chosen for the LBB analysis because they have the highest stresses with the most limiting material properties of both accumulator A and B lines.

Response: (a) Figure 1-2 of SIR-99-036 depicts the connection of Accumulator B piping to the Cold Leg. Check Valve 867B isolates the rest of the Accumulator B piping from the reactor coolant system (RCS). The piping within Nodes 60 and 80 is the unisolable part of the piping and consequently included in the LBB evaluation.

(b) As shown in Figure 1-1 of SIR-99-036, only Node 856 of the Accumulator A piping is included in the LBB analysis since the other nodes were included in the LBB evaluation of the Residual Heat Removal (RHR) which has been approved by the NRC in a SER dated February 25, 1999. Per Section 4.2 of SIR-99-036, only the welds and no pipe segments in both Accumulator A and B piping were considered.

(c) The nodes that are considered in the LBB evaluation of Accumulator A and B piping are located at welds in a tee and an elbow and consequently reflect high stresses due to the stress intensification effects. Section 4.2 of SIR-99-036 documents the expectation that the SMAW weld properties will provide the most conservative critical flaw and leakage flaw sizes. Weld property values were taken from the EPRI Ductile Fracture Handbook (1989), which was also the basis for flaw acceptance criteria in the ASME Code (Section XI).

10. Section 4.2 discusses the material properties for the accumulator piping and welds. Discuss whether there are any Alloy 82/182 welds located in the accumulator lines that are covered by the LBB application.

Response: Review of the original construction weld procedure used in welding piping for Accumulator A and B systems revealed that stainless steel electrodes (E316) were utilized for generating GTAW and SMAW passes. There are no Alloy 82/182 welds present anywhere in the Accumulator A and B piping system.

ATTACHMENT 2

R. E. GINNA NUCLEAR POWER PLANT

DESCRIPTION AND ASSESSMENT OF PROPOSED CHANGE

Subject: **Modify licensing basis for Ginna Station to incorporate "leak-before-break" per GDC 4 for the accumulator lines, and the pressurizer surge line accounting for thermal stratification.**

1.0 DESCRIPTION

2.0 PROPOSED CHANGE

3.0 BACKGROUND

4.0 TECHNICAL ANALYSIS

5. NO SIGNIFICANT HAZARDS CONSIDERATION

6.0 ENVIRONMENTAL CONSIDERATION

7.0 REFERENCES

1.0 DESCRIPTION

The proposed change would modify the Ginna Station licensing basis, so that, in accordance with General Design Criterion 4, "leak-before-break" could be applied to the accumulator "A" and "B" lines, as well as the pressurizer surge line.

The detailed analyses were provided to the NRC in a submittal dated September 30, 2004, "Fracture Mechanics Analysis per GDC-4".

It is requested that this change be approved prior to September 30, 2005, so that it could be incorporated into Revision 19 of the Ginna Station UFSAR.

2.0 PROPOSED CHANGES

There are no changes proposed to the Ginna Operating License or Technical Specifications. Ginna is requesting this change to the licensing basis so that all branch connections to the Reactor Coolant System would have been appropriately analyzed for "leak-before-break". The Residual Heat Removal inlet and outlet lines had previously been approved by SER dated February 25, 1999.

3.0 BACKGROUND

As part of the Systematic Evaluation Program, SEP Topic III-5.A "High Energy Line Breaks Inside Containment", it was determined that a fracture mechanics evaluation was to be performed for the pressurizer surge line as well as accumulator surge line "A".

The acceptance of this analysis was issued in an SER dated June 28, 1983, IPSAR Section 4.13, "Effects of Pipe Break on Structures, Systems, and Components Inside Containment for the R. E. Ginna Nuclear Power Plant".

Because this analysis did not account for the effects of thermal stratification on the pressurizer surge line, it was decided to resubmit this analysis. WCAP 16311-P, Rev. 0 was submitted by letter dated September 30, 2004.

Since only accumulator line "A" was covered by the original 6/28/83 SER, it was further decided to perform the fracture mechanics analysis for accumulator line "B", and reanalyze accumulator line "A". These analyses were also provided in the September 30, 2004 submittal.

4.0 TECHNICAL ANALYSIS

The technical analysis of these lines was submitted on September 30, 2004.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Ginna LLC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment", as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previous evaluated?

Response: No

The proposed changes use an approved fracture mechanics methodology, in accordance with 10 CFR 50, Appendix A, GDC-4 to demonstrate that the probability of fluid system rupture for these lines attached to the Reactor Coolant System is extremely low under conditions associated with the design basis for the piping.

The proposed changes do not adversely affect accident initiators or precursors nor significantly alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupation/public radiation exposures. The proposed changes do not affect the probability of an accident occurring since they reflect a change in plant design basis that is consistent with current Regulations. The proposed changes cannot increase the consequences of postulated accidents since LOCA and methods containment analysis will not be changed. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change does not create the possibility of a new or different kind of accident, since it simply provides an analytical justification for demonstrating that the probability of a fluid system rupture is extremely small. Leak-before-break justifications per GDC-4 still require that ECCS, containment, and EQ requirements be maintained consistent with the original postulated accident assumptions – only protection from dynamic effects is modified.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

The proposed changes apply very conservative approved analytical methods to demonstrate that the probability of a fluid system rupture is very low. This analysis justifies differences in protection from dynamic effect is associated with these extremely low probability ruptures. For overall ECCS, containment, and EQ requirements, there will be no changes to the licensing basis.

6.0 ENVIRONMENT CONSIDERATION

Ginna LLC has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration; and

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite since there is no change in the type or quantities of material available for release than that previously analyzed; and

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure since the change in plant configuration does not significantly increase overall operations and maintenance requirements nor is any different type of equipment required to be installed.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

7.0 REFERENCES

7.1 Letter from Joseph A. Widay to Robert L. Clark, "Fracture Mechanics Analysis per GDC-4", September 30, 2004.

7.2 Letter from Guy Vissing (NRC) to Robert C. Mecredy (RG&E), "Staff Review of the Submittal by Rochester Gas and Electric Company to Apply Leak-Before-Break Status to Portions of the R.E. Ginna Nuclear Power Plant Residual Heat Removal Piping", February 25, 1999.

7.3 Letter from Dennis M. Crutchfield (NRC) to John E. Maier (RG&E), "IPSAR Section 4.13, "Effects of Pipe Break on Structures, Systems, and Components in Containment for the R.E. Ginna Nuclear Power Plant", June 28, 1983.