SAFETY EVALUATION REPORT
OFFICE OF NUCLEAR REACTOR REGULATION
STEAM GENERATOR SNUBBER REPLACEMENT PROGRAM
R. E. GINNA NUCLEAR POWER PLANT
DOCKET NO.: 50-244
MECHANICAL ENGINEERING BRANCH
DIVISION OF ENGINEERING AND SYSTEMS TECHNOLOGY

1.0 BACKGROUND

On September 22, 1987, RG&E notified the NRC (Reference 1) that it intends to replace six of the eight hydraulic snubbers per steam generator with bumpers (rigid structural members) at the R. E. Ginna Nuclear Power Plant. The purpose of this replacement is to reduce required maintenance activities and aid in keeping radiation exposures as-low-as-reasonable-achievable.

Reference 1 also indicated that the analysis which will provide justification for the snubber replacement will involve the application of updated pipe break design criteria (leak-before-break) to reduce maximum loads on the reactor coolant loop system and its supports, and the change in MEB BTP 3-1 to eliminate arbitrary intermediate breaks in the main steam and feedwater lines.

RG&E intended to perform this replacement as a plant modification subject to review per 10CFR50.59. After evaluation of the technical material in Reference 1 and communication with the licensee, the EMEB staff concluded that the replacement of these snubbers would require a licensing amendment. This conclusion was based on the final narrow-scope rule on the amendment to GDC-4 as published in the Federal Register Notice of April 11, 1986, (p. 12504), which stated that changing heavy-component-support load-path designs would involve an unreviewed safety question, and would thus require a licensing amendment.

On May 13, 1988, RG&E submitted an Application for Amendment to the Operating License (Reference 2) to reflect replacement of steam generator snubbers with rigid structural members. Included in Reference 2 are also a report summarizing the safety evaluation analyses which were performed to demonstrate no adverse consequences as a result of the snubber replacement, and responses to a series of questions from the EMEB staff which were transmitted to RG&E on April 13, 1988 (Reference 3).

2.0 EVALUATION

2.1 ANALYSIS

The upper portion of each of the two steam generators (SG) at R. E. Ginna are currently restrained against lateral seismic and pipe break loads by eight large capacity hydraulic snubbers. The snubbers are installed in four pairs, with one pair approximately parallel to the hot leg. The redesigned SG upper support configuration will retain this pair of snubbers while the other six will be replaced with bumpers (rigid structural members). Each of these bumpers consists of a structural assembly which is rigid in compression and permits freedom of movement in the tensile direction.

The size and design of these bumpers were based on the primary loop qualification performed by the NSSS vendor, Westinghouse Electric Corporation (WEC). The reactor coolant system (RCS) with the modified SG upper lateral support

configurations was analyzed for dead weight, internal pressure, thermal expansion, seismic events (OBE & SSE), postulated pipe ruptures at the main steam and feedwater nozzles and postulated pipe ruptures at the pressurizer surge, SI accumulator and RHR auxiliary line nozzles. No pipe breaks or asymmetric blow-down loads were postulated in the RCS piping itself, since an exemption to GDC-4 of Appendix A, 10CFR50 was granted to R. E. Ginna by letter dated September 9, 1986 (Reference 4).

The postulated pipe break locations in the main steam and feedwater systems were reviewed in accordance with NRC Generic Letter 87-11 (Reference 5). Based on the provisions of this letter RG&E determined that the terminal end pipe break postulated at the steam generator feedwater inlet nozzle defined the limiting pipe break design basis load for the SG upper lateral support system under faulted conditions.

The static, thermal and seismic analyses of the RCS were performed using a two-loop model to obtain component and support loads and displacements. The seismic analysis was performed using the envelope response spectrum method with uniform modal damping equal to 2% (OBE) and 4% (SSE) of critical damping. These damping values were approved by the NRC (AEC) for use in WEC RCS analyses by letter dated May 16, 1974 (Reference 6). The new SG upper lateral supports designs were represented by members with different stiffnesses in tension and compression,

but were modeled by members with the average stiffness in the analyses. Likewise, the existing tubular support columns under the steam generators and reactor coolant pumps were modeled by members with the average of the stiffnesses in tension and compression. The staff reviewed these linear approximations to essentially non-linear members, and concluded that they were appropriate for the analysis of the R. E. Ginna RCS on the basis that substantial margin was calculated in these members with respect to their load capacities. However, the staff intends to seek, on a generic basis, additional clarification on this subject from Westinghouse since this approach was also used in other Westinghouse plants.

The pipe-rupture analyses were performed on a one-loop model using a dynamic time history method. Pipe breaks were postulated at the loop branch connections of the pressurizer surge, RHR and SI accumulator piping systems. The transient forcing functions representing thrust and jet impingement were applied to the RCS analytical model at the lumped mass points where each auxiliary line joins the RCS. The blowdown forcing functions which represent traveling compression blowdown waves within the RCS were also applied to the model at each change in piping direction or change in flow area. The forcing functions due to the pipe breaks postulated at the main steam outlet nozzle and the feedwater inlet nozzle are applied at these locations on the SG.

All analyses of the RCS were performed using the proprietary WEC computer program "WESTDYN". This program performs both response spectrum analysis and time history analysis. The analyses of the equipment supports were performed using the widely used structural computer program "STRUDL", of which WEC has a proprietary version. The basis and application of these programs was reviewed and accepted by the staff in 1981 (Reference 4).

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The piping stresses for the normal, upset, emergency and faulted plant conditions were evaluated using the requirements of ANSI B31.1, 1967 Edition (Reference 8). The design of the bumpers and equipment supports was based on requirements of the current edition of the original design code, AISC Manual of Steel Construction, (8th Edition), and the provisions of the ASME B&PV Code Section III, Subsection NF and Appendix F, 1974 Edition.

2.2 RESULTS

The results of the analyses show that the maximum stresses in the RCS piping with the redesigned steam generator lateral support configuration are within the appropriate design stress limits for the various plant conditions. Likewise, the stresses and loads in the RCS heavy equipment nozzles and supports were also shown to have adequate margin as compared to the corresponding design limits.

With the redesigned steam generator upper lateral support configuration, the analysis generated revised forces and moments in the RCS. These revised loads were compared with the corresponding loads for this plant in Generic Letter 84-04. This comparison verified that the leak-before-break conclusions for Ginna in WCAP-9558, Rev. 1, (Reference 9), remain valid for the redesigned support configuration. The stresses in the main steam lines were evaluated, and the maximum stress intensity at all intermediate locations between terminal ends for combined pressure, dead weight, thermal and OBE loading was shown to be less than the threshold stress intensity specified in GL 87-11, thus providing justification for not specifying intermediate break locations in the main steam lines inside containment.

Prior to this stress evaluation, the controlling design load for the SG upper lateral support system was an arbitrary intermediate pipe break in the horizontal main steam line near the top of the SG. With this calculation, the loading on the SG due to a main steam postulated break at the terminal end changed to the vertical direction along the axis of the SG. Similar considerations are also applicable to the feedwater lines, in which case the previously defined loads due to a postulated terminal-end break at the feedwater inlet nozzle became the limiting pipe break design basis load under faulted conditions.

WEC also performed an evaluation of the building structures which provide the anchors for the bumpers and the snubbers, and were found to be adequate for the design loads associated with the new SG upper support configurations.

The maximum piping stresses in the RCS in the cold shut-down condition under dead weight pressure, and seismic loading (SSE) were found to be well within the allowable stress (emergency condition). Likewise, all loads on the primary equipment supports were found to be well within the capacity for the corresponding support components.

3.0 CONCLUSION

Based on the review of the analyses and results provided the staff finds that RG&E has shown that the RCS and the heavy component supports with the proposed steam generator support configuration meets the design code limits specified in the FSAR when designed against the specified design basis loads. The intended modifications are therefore found to be acceptable as well as the proposed modifications to Section 3.13 of the Technical Specifications in Appendix A of the facility operating license. The staff also finds that the licensee has shown in a reasonable manner that this change will not result in a significant increase in the probability or consequences of an accident previously evaluated, it will not create the possibility of a new or different kind of accident from any accident previously evaluated and does not involve a significant reduction in a margin of safety.

4.0 REFERENCES

1. September 22, 1987, letter from R. W. Kober, (RG&E) to C. Stahle (NRC), "Steam Generator Snubber Replacement Program at the R. E. Ginna Nuclear Power Plant," with attachment.
2. May 13, 1988, letter from B. A. Snow, (RG&E) to C. Stahle (NRC), with attached report "Steam Generator Hydraulic Snubber Replacement Program", Revision 2, dated May 8, 1988.
3. April 13, 1988, letter from C. Stahle, (NRC) to R. W. Kober (RG&E) "Snubber Replacement Program at Ginna".
4. September 9, 1987, letter from D. C. DiIanni, (NRC) to R. W. Kober (RG&E).
5. June 19, 1987, letter from F. J. Miraglia, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements (Generic Letter 87-11)".
6. May 16, 1964, letter from D. B. Vassallo, (AEC) to R. Salvatori, (Westinghouse Electric Corporation).
7. April 7, 1981, letter from R. L. Tedesco, (NRC) to T. M. Anderson, (WEC) on the approval of WCAP-8252.
8. ANSI B31.1 Power Piping Code 1967 Edition, including Summer 1973 Addenda.
9. WCAP-9558, Rev. 1, "Mechanical Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Thin Wall Crack", June 1980.

SALP REPORT

PLANT: R. E. Ginna Nuclear Plamt
DOCKET NO.: 50-244

LICENSEE: Rochester Gas & Electric
REVIEWER: M. Hartzman
LICENSING ACTIVITY: SER

EVALUATION CRITERIA	RATING	REMARKS
1. Management Involvement and Control in Assuring Quality	2	RG&E has been fully involved in this effort
2. Approach to Resolution of Technical Issues from a safety standpoint	2	There are no unresolved concerns in the submittal by RG&E
3. Responsiveness to NRC Initiatives	3	
4. Enforcement History	N/A	
5. Reporting and Analysis of Reportable Events	N/A	
6. Staffing	N/A	

The summary SALP rating for this submittal is 2.