Attachment 2

EGG-MS-8923

TECHNICAL EVALUATION REPORT ON THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN: ROCHESTER GAS AND ELECTRIC CORPORATION, R. E. GINNA NUCLEAR POWER STATION, DOCKET NUMBER 50-244

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ABSTRACT

This report presents the results of the evaluation of the R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection (ISI) Program Plan, submitted July 21, 1989, including the requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements which the Licensee has determined to be impractical. The R. E. Ginna Nuclear Power Station Third 10-Year Interval ISI Program Plan is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during previous Nuclear Regulatory Commission (NRC) reviews. The requests for relief are evaluated in Section 3 of this report.

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Operating Reactor Licensing Issues Program,
Review of ISI for ASME Code Class 1, 2, and 3 Components

SUMMARY

The Licensee, Rochester Gas and Electric Corporation, has prepared the R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection (ISI) Program Plan to meet the requirements of the 1986 Edition of the ASME Code Section XI. The third 10-year interval began January 1, 1990 and ends December 31, 1999.

The information in the R. E. Ginna Nuclear Power Station Third 10-Year Interval ISI Program Plan, submitted July 21, 1989, was reviewed. Included in the review were the requests for relief from the ASME Code Section XI requirements which the Licensee has determined to be impractical. As a result of this review, a request for additional information (RAI) was prepared describing the information and/or clarification required from the Licensee in order to complete the review. The Licensee provided additional information in the submittal dated January 16, 1990.

Based on the review of the R. E. Ginna Nuclear Power Station Third 10-Year Interval ISI Program Plan, the Licensee's response to the Nuclear Regulatory Commission's RAI, and the recommendations for granting relief from the ISI examination requirements that have been determined to be impractical, it is concluded that the R. E. Ginna Nuclear Power Station Third 10-Year Interval ISI Program Plan, with the exception of Request for Relief Nos. 10 and 13, is acceptable and in compliance with 10 CFR 50.55a(g)(4).

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TECHNICAL EVALUATION REPORT ON THE
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1. INTRODUCTION

Throughout the service life of a water-cooled nuclear power facility, 10 CFR 50.55a(g)(4) (Reference 1) requires that components (including supports) that are classified as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, Class 2, and Class 3 meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Station Components," (Reference 2) to the extent practical within the limitations of design, geometry, and materials of construction of the components. This section of the regulations also requires that inservice examinations of components and system pressure tests conducted during successive 120-month inspection intervals shall comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed therein. The components (including supports) may meet requirements set forth in subsequent editions and addenda of this Code which are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The Licensee, Rochester Gas and Electric Corporation, has prepared the R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection (ISI) Program Plan to meet the requirements of the 1986 Edition of the ASME Code Section XI. The third 10-year interval began January 1, 1990 and ends December 31, 1999.

As required by 10 CFR 50.55a(g)(5), if the licensee determines that certain Code examination requirements are impractical and requests relief from them, the licensee shall submit information and justifications to the Nuclear Regulatory Commission (NRC) to support that determination.

Pursuant to 10 CFR 50.55a(g)(6), the NRC will evaluate the licensee's determinations under 10 CFR 50.55a(g)(5) that Code requirements are impractical. The NRC may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

The information in the R. E. Ginna Nuclear Power Station Third 10-Year Interval ISI Program Plan (Reference 3), submitted July 21, 1989, was reviewed, including the requests for relief from the ASME Code Section XI requirements which the Licensee has determined to be impractical. Review of the ISI Program Plan was performed using the Standard Review Plans of NUREG-0800 (Reference 4), Section 5.2.4, "Reactor Coolant Boundary Inservice Inspections and Testing," and Section 6.6, "Inservice Inspection of Class 2 and 3 Components."

In a letter dated November 16, 1989 (Reference 5), the NRC requested additional information required in order to complete the review of the ISI Program Plan. Additional information was provided by the Licensee in a submittal dated January 16, 1990 (Reference 6).

The R. E. Ginna Nuclear Power Station Third 10-Year Interval ISI Program Plan is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during previous NRC reviews.

The requests for relief are evaluated in Section 3 of this report. Unless otherwise stated, references to the Code refer to the ASME Code, Section XI, 1986 Edition. Specific inservice test (IST) programs for pumps and valves are being evaluated in other reports.

2. EVALUATION OF INSERVICE INSPECTION PROGRAM PLAN

This evaluation consisted of a review of the applicable program documents to determine whether or not they are in compliance with the Code requirements and any license conditions pertinent to ISI activities. This section describes the submittals reviewed and the results of the review.

2.1 Documents Evaluated

Review has been completed on the following information provided by the Licensee:

(a) R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection Program Plan; and

(b) Letter, dated January 16, 1990, containing the Licensee's response to the NRC request for additional information.

2.2 Compliance with Code Requirements

2.2.1 Compliance with Applicable Code Editions

The Inservice Inspection Program Plan shall be based on the Code editions defined in 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(b). Based on the starting date of January 1, 1990, for the R. E. Ginna Nuclear Power Station, the Code applicable to the third 10-year interval ISI program plan is the 1986 Edition. As stated in Section 1 of this report, the Licensee has written the R. E. Ginna Nuclear Power Station Third 10-Year Interval ISI Program Plan to meet the requirements of the 1986 Edition of the ASME Code.

2.2.2 Acceptability of the Examination Sample

Inservice volumetric, surface, and visual examinations shall be performed on ASME Code Class 1, 2, and 3 components and their supports using sampling schedules described in Section XI of the ASME Code and

10 CFR 50.55a(b). Sample size and weld selection have been implemented in accordance with the Code and appear to be correct.

2.2.3 Exclusion Criteria

The criteria used to exclude components from examination shall be consistent with Paragraphs IWB-1220, IWC-1220, IWC-1230, IWD-1220, and 10 CFR 50.55a(b). The exclusion criteria have been applied by the Licensee in accordance with the Code as discussed in the ISI Program Plan and appear to be correct.

2.2.4 Augmented Examination Commitments

In addition to the requirements specified in Section XI of the ASME Code, the Licensee has committed to perform augmented examinations in accordance with NRC Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1, (Reference 7) and NRC Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," Revision 1 (Reference 8).

Also, circumferential butt welds in high energy piping outside of containment where a failure would result in unacceptable consequences will receive radiographic or ultrasonic volumetric examinations and magnetic particle surface examinations.

2.3 Conclusions

Based on review of the documents listed above, it is concluded that the R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection Program Plan is acceptable and in compliance with 10 CFR 50.55a(g)(4).

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3. EVALUATION OF RELIEF REQUESTS

Requests for relief from the ASME Code requirements that the Licensee has determined to be impractical for the third 10-year inspection interval are evaluated in the following sections.

3.1 Class 1 Components

3.1.1 Reactor Pressure Vessel

3.1.1.1 Request for Relief No. 1, Examination Category B-A, Item B1.30, Deferral of RPV Shell-to-Flange Weld Examinations

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.30 requires a 100% volumetric examination of the Reactor Pressure Vessel shell-to-flange weld as defined by Figure IWB-2500-4. Note 5 states that, if partial examinations are conducted from the flange face, the remaining volumetric examinations required to be conducted from the vessel wall may be performed at or near the end of each inspection interval. Note 6 states that the examination of shell-to-flange welds may be performed during the first and third inspection periods in conjunction with the nozzle examination of Examination Category B-D. At least 50% of shell-to-flange welds shall be examined by the end of the first inspection period and the remainder by the end of the third inspection period.

Licensee's Code Relief Request: Relief is requested to defer the volumetric examination of the Reactor Pressure Vessel (RPV) shell-to flange weld to the end of the inspection interval.

Licensee's Proposed Alternative Examination: The Licensee states that, during the third interval, 100% of the accessible length of all RPV welds, including the shell-to-flange weld, will be performed at or near the end of the interval when all the required examinations can be performed at the same time.

<u>Licensee's Basis for Requesting Relief</u>: The Licensee states that the required shell-to-flange examination is impractical if performed during the periods specified as it can be accomplished only from the flange surface.

During the first two inspection intervals, 100% of the accessible length of the RPV welds, including the shell-to-flange weld, was examined at or near the end of the interval when the entire examination could be performed from both the flange surface and the vessel wall. This is a more practical approach in that the required examinations from both surfaces can be performed at the same time.

The Licensee's January 16, 1990 submittal stated that Rochester Gas and Electric received relief for R. E. Ginna for the second interval (1979-1989) allowing deferral of all vessel mechanized examinations including category B-A to the end of the second interval. Full examination of the RPV shell-to-flange weld (Category B-A) was performed from the flange surface and from the vessel shell during the end of interval March-April 1989 refueling outage. The examination was performed in accordance with the 1974 Edition, Summer 1975 Addenda of the ASME Code.

Indications were recorded and analyzed during the series of examinations performed at the 1989 outage. Comparisons of their size to the same indications recorded during the preservice examination and the 1979 end of the first interval inservice examination led to no indications of growth of defects. The indications were sized utilizing both conventional beam spread techniques and specially developed focused beam sizing techniques. No evidence of growth was detected by utilizing these examination techniques. These examinations were witnessed by the NRC and their consultants during the 1989 outage.

[illegible]

The results of the 1989 examinations with comparative sizing calculations and fracture mechanics evaluation were submitted to the NRC on May 4, 1989. The data provided were accepted by the NRC on June 5, 1989 with no requirements for augmented examinations. The Licensee has, therefore, scheduled the next series of Category B-A RPV examinations for the 1999 time frame (end of this 10-year interval). Examination Category B-A in Table IWB-2500 Item number B1.30 already allows examination deferral as indicated in footnote 5 of the notes. By following the 1986 Code criteria, the Code would require a vessel-to-flange examination from the flange seal surface during the first period of the third interval. Since all reactor vessel examinations, including the vessel-to-flange weld from the flange surface and wall, were accomplished in 1989, the prescribed B-A, B1.30, Code criteria would allow another 10 years before the reexamination.

Since the examination of the RPV vessel-to-flange weld was accomplished completely in 1989, the Licensee proposes to reexamine the weld at the end of the third interval. The Licensee also believes that the results obtained from an examination performed during the first period of the third interval would not provide the best data to ensure additional safety and system reliability. Also, performing this type of examination at a constant frequency rather than a year or two following a major examination of that same component will produce more meaningful results.

<u>Evaluation</u>: Paragraph IWB-2420(a), "Successive Inspections," states that the sequence of component examinations established during the first inspection interval shall be repeated during each successive inspection interval, to the extent practical. Since the examinations were performed during the third period of the second interval with NRC approval, the intent of the Code will be met by examining these welds during the third period of the third interval.

Conclusions: Based on the above, it is concluded that the intent of the Code will be met by the alternative schedule. Therefore, it is recommended that relief be granted as requested.

3.1.1.2 Request for Relief No. 2, Examination Category B-D, Items B3.90 and B3.100, Reactor Pressure Vessel Nozzle-to-Vessel Welds and Nozzle Inside Radius Sections

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.90 requires a 100% volumetric examination of the RPV nozzle-to-vessel welds as defined by Figure IWB-2500-7. Inspections may be partially deferred under the following conditions: If examinations are conducted from inside the component and the nozzle weld is examined by straight beam ultrasonic method from the nozzle bore, the remaining examinations required to be conducted from the shell inside diameter may be performed at or near the end of each inspection interval.

Examination Category B-D, Item B3.100 requires a 100% volumetric examination of the RPV nozzle inside radius sections as defined by Figure IWB-2500-7. At least 25% but not more than 50% (credited) of the nozzles shall be examined by the end of the first inspection period and the remainder by the end of the inspection interval.

Licensee's Code Relief Request: Relief is requested to defer the volumetric examinations of the RPV nozzle-to-vessel welds and nozzle inside radius sections to the end of the inspection interval.

Licensee's Proposed Alternative Examination: The Licensee proposes to perform both nozzle-to-vessel examinations (from the nozzle bore and from the shell inside diameter) at or near

the end of the interval. The nozzle inside radius examinations will also be performed at this time. This more practical approach will allow all the required examinations to be performed at the same time on all the nozzles and nozzle inside radii.

Licensee's Basis for Requesting Relief: The Licensee states that examinations from the nozzle bore and nozzle inside radius examinations can be performed on only two (outlets) of the six major nozzles without removal of the core barrel. The mechanized examination of the two accessible nozzle and inside radius sections is expensive, and the nozzle-to-vessel examination is only a partial examination from the nozzle bore. From a technical position considering the progress which is being made in ultrasonic examination equipment and techniques and for the correlation of data obtained from the bore with that obtained from the shell, it is highly desirable to perform both examinations at the same time.

The Licensee's January 16, 1990 submittal stated that Rochester Gas and Electric received relief for R. E. Ginna for the second interval (1979-1989) allowing deferral of all vessel mechanized examinations including category B-D to the end of the second interval. Full examinations of the RPV nozzle-to-vessel welds (Category B-D) were performed from the nozzle bores during the end of interval March-April 1989 refueling outage. All RPV nozzle inside radius section examinations were also performed at this time. All examinations were performed in accordance with the 1974 Edition, Summer 1975 Addenda of the ASME Code.

Indications were recorded and analyzed during the series of examinations performed at the 1989 outage. Comparisons of their size to the same indications recorded during the preservice examination and the 1979 end of the first interval inservice examination led to no indications of growth of defects. The indications were sized utilizing both

conventional beam spread techniques and specially developed focused beam sizing techniques. No evidence of growth was detected by utilizing these examination techniques. These examinations were witnessed by the NRC and their consultants during the 1989 outage.

Results of the 1989 examinations with comparative sizing calculations and fracture mechanics evaluation were submitted to the NRC on May 4, 1989. The data provided were accepted by the NRC on June 5, 1989 with no requirements for augmented examinations. The Licensee has, therefore, scheduled the next series of Category B-D RPV examinations for the 1999 time frame (end of this 10-year interval). The Category B-D welds and inside radius sections for all vessels other than the reactor vessel have been scheduled for examination during the three periods of the third interval. By following the 1986 Code criteria, the Code would require a nozzle examination from the bore during the first period of the third interval. Since all reactor vessel examinations, including nozzle-to-shell from both the bore and shell and nozzle inside radius sections, were accomplished in 1989, the prescribed B-D, B3.90 and B3.100 Code criteria would allow another 10 years before the reexamination.

Since the examinations were accomplished completely in 1989, the Licensee proposes to reexamine them at the end of the third interval. The Licensee also believes that the results obtained from an examination performed during the first period of the third interval would not provide the best data to ensure additional safety and system reliability. Also, performing this type of examination at a constant frequency rather than a year or two following a major examination of that same component will produce more meaningful results.

<u>Evaluation</u>: Paragraph IWB-2420(a), "Successive Inspections," states that the sequence of component examinations established during the first inspection interval shall be repeated during

each successive inspection interval, to the extent practical. Since the examinations were performed during the third period of the second interval with NRC approval, the intent of the Code will be met by examining these welds during the third period of the third interval.

Conclusions: Based on the above, it is concluded that the intent of the Code will be met by the alternative schedule. Therefore, it is recommended that relief be granted as requested.

3.1.2 Pressurizer (No relief requests)

3.1.3 Heat Exchangers and Steam Generators (No relief requests)

3.1.4 Piping Pressure Boundary

3.1.4.1 Request for Relief No. 16 (Part 1 of 2), Examination Category B-K-1, Item B10.10, Class 1 Piping Integrally Welded Attachments

NOTE: Request for Relief. No. 16 was withdrawn by the Licensee in the January 16, 1990 submittal, which states the following:

> "Relief Request Number 16 is not required by the Code. The associated program plan tables will be corrected and updated to assure accurate accounting. This activity will be performed to reflect the 'non-requirement' code status since the associated thicknesses are < 5/8" for class 1 and < 3/4" for class 2. Relief Request Number 16 is not required and will be deleted."

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3.1.5 Pump Pressure Boundary

3.1.5.1 Request for Relief No. 4, Examination Category B-L-1 and B-L-2, Items B12.10 and B12.20, Class 1 Pump Casing Welds and Pump Casings

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-L-1, Item B12.10 requires a 100% volumetric examination of Class 1 pump casing welds as defined by Figure IWB-2500-16. Examination Category B-L-2, Item B12.20 requires a 100% visual (VT-3) examination of the internal surfaces of Class 1 pump casings.

Licensee's Code Relief Request: Relief is requested from performing the Code-required volumetric and visual (VT-3) examinations of the reactor coolant pump casing welds and internal surfaces, respectively.

Licensee's Proposed Alternative Examination: A visual examination (VT-1) of the external surfaces of the welds of one pump casing will be performed. A visual examination (VT-3) of the internal surfaces will be performed each time a pump is disassembled. A visual examination (VT-2) of the exterior of all pumps will be performed during the hydrostatic pressure test required by Table IWB-2500-1, Category B-P. Also, an evaluation will be performed to demonstrate the safety and serviceability of the pump casing. In making this assessment, thermal aging embrittlement and any other processes that may degrade the properties of the pump casing during service will be considered.

Licensee's Basis for Requesting Relief: The two reactor coolant pumps for R. E. Ginna are Westinghouse Model 93 pumps. Each pump casing is fabricated by welding four stainless steel (SA351 CF8) castings together. Thus, there are three

circumferential pressure retaining welds requiring volumetric inspection in accordance with Category B-L-1.

The Licensee states that the unsuitability of ultrasonic examination was demonstrated during the "A" reactor coolant pump examination in 1980 when an attempt was made to determine the wall thicknesses using ultrasonic examination techniques. At that time, it was determined that the casing welds must be inspected using the miniature linear accelerator (MINAC).

Radiographic examination using the MINAC was performed on the R. E. Ginna "A" reactor coolant pump during the Spring 1981 refueling outage. This examination was performed by placing the MINAC inside the pump casing and placing the film on the outside of the pump. To perform the examination, the pump was completely disassembled. Also, the insulation on the casing exterior was removed to permit film placement.

Additionally, the pump bowl must be dry for installation of the MINAC. Therefore, all fuel assemblies were removed from the reactor vessel and the vessel water level lowered to below the nozzles. Complete disassembly of the pump was also required to conduct the VT-1 examination in accordance with Category B-L-2.

No problems have been found with the welds at R. E. Ginna or other sites. Additionally, no problems have been found during the Category B-L-2 visual examination. Visual examination was conducted at R. E. Ginna by using the video camera on the MINAC.

The whole body exposure to personnel during the spring 1981 refueling outage directly attributable to the reactor coolant pump "A" examinations is 93,067 millirem. This does not include the dose received during the complete core unload to get the plant in condition for the reactor coolant pump disassembly.

The nuclear industry has been successfully applying leak-before-break concepts to primary loop and Class 1 auxiliary piping systems of commercial nuclear power plants. Currently, the analyses supporting such concepts come under the review of the NRC by General Design Criteria-4 (GDC-4).

Eight different models of reactor coolant pumps are used in Westinghouse PWRs. Model 93 methodology used in the analyses is consistent with that recommended in NUREG-1061, Vol. 3 and GDC-4. A finite element stress analysis model for the Model 93 pump was developed.

The reactor coolant pump casings are cast stainless steel. The chemistries of each heat of material used in the pumps were used to determine the fracture toughness. The phenomenon of thermal aging was addressed.

The program successfully demonstrates that leak-before-break analyses are applicable to all primary pump casings of all Westinghouse design PWRs for which the screening loads are reasonably applicable and the fracture toughness is known.

<u>Evaluation</u>: The volumetric examination is impractical to perform to the extent required by the Code because radiographic techniques for the subject pump casing welds require complete disassembly of the pumps, and ultrasonic examinations are limited by the coarse grain structure inherent in thick stainless steel castings.

The visual examination is performed to determine whether unanticipated severe degradation of the casing is occurring due to phenomena such as erosion, corrosion, or cracking. However, previous experience during examination of similar pumps at other plants has not shown any significant degradation of pump casings.

The disassembly of the pumps for the sole purpose of inspection is a major effort and, in addition to the possibility of damage to the pumps, could result in personnel receiving excessive radiation exposure. However, if the pumps are disassembled for maintenance, the Code-required volumetric and visual examinations should be performed.

Conclusions: Based on the above evaluation, it is concluded that the Code requirement is impractical. Therefore, it is recommended that relief be granted provided that (a) visual examination (VT-3) of the internal surfaces of the pumps is performed whenever the pumps are disassembled, (b) volumetric examination of the pump casing welds is performed whenever the welds are exposed due to disassembly of the pump, (c) a surface examination of the pump casing welds is performed if the pumps are not disassembled, and (d) if the pumps have not been disassembled, this fact should be reported by the Licensee in the ISI Summary Report at the end of the interval.

3.1.6 Valve Pressure Boundary

3.1.6.1 Request for Relief No. 5, Examination Category B-M-2, Item B12.50, Class 1 Valve Body Internal Surfaces

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-M-2, Item B12.50 requires a 100% visual (VT-3) examination of Class 1 valve body internal surfaces. Examinations are limited to at least one valve within each group of valves that are of the same size, constructional design, and manufacturing method, and that perform similar functions in the system.

Licensee's Code Relief Request: Relief is requested from performing the Code-required visual (VT-3) examination of the valve body internal surfaces of one valve in each of the following four groups of Class 1 valves:

Size (In.)	Valve Number	Mfg/Type	Line Number
10	842A	Darling/Check	10-SI2-2501-A
10	842B	Darling/Check	10A-SI2-2501-B
10	867A	Darling/Check	10-SI2-2501-B
10	867B	Darling/Check	10A-SI2-2501-B
10	700	Velan/Gate	10A-RCO-2501-A
10	701	Velan/Gate	10A-AC7-2501-A
10	720	Velan/Gate	10A-AC7-2501-B
10	721	Velan/Gate	10A-AC7-2501-B
6	853A	Velan/Gate	6A-RC-2501-A
6	853B	Velan/Gate	6A-RC-2501-B
6	852A	Velan/Check	6A-RC-2501-A
6	852B	Velan/Check	6A-RC-2501-B

Licensee's Proposed Alternative Examination: The Licensee states that, as standard maintenance practice dictates, when these valves are disassembled for maintenance purposes, a visual examination of the internals and internal pressure boundary surfaces will be performed to the extent practical.

Licensee's Basis for Requesting Relief: The Licensee states that, to complete the subject examination, unnecessary expenditures of man-hours and manrem are required with essentially no compensating increase in plant safety. Also, the structural integrity afforded by valve casing material used will not significantly degrade over the lifetime of the valve.

Based on data compiled from a plant similar in age and design to Ginna Station, it is expected that approximately 100 manhours and 5 manrem exposure would be required to disassemble, inspect, and reassemble these valves. Performing this visual examination under such adverse conditions, high dose rates (30 to 40 R/hr), and poor as-cast surface conditions, realistically provides little additional information as to the valve's casing integrity.

The valve material, a high-strength cast stainless steel (ASTM A351-CF8), is widely used in the nuclear industry and has

performed extremely well. The presence of some delta ferrite (typically 5% or more) substantially increases resistance to intergranular stress corrosion cracking. The delta ferrite also helps the material to resist pitting corrosion in chloride containing environments.

The Licensee further states that adequate safety margins are inherent in the basic valve design and that public health and safety will not be adversely affected by not performing a visual examination of the valve internal pressure boundary surfaces. Additionally, this visual examination adds little or no value to the overall safety of the plant and subjects plant personnel to unnecessary radiation exposure.

Evaluation: The visual examination is performed to determine whether unanticipated severe degradation of the valve body is occurring due to phenomena such as erosion, corrosion, or cracking. However, previous experience during examination of similar valves at other plants has not shown any significant degradation of valve bodies.

Disassembly of the valves for the sole purpose of inspection is a major effort and, in addition to the possibility of damage to the valves, could result in personnel receiving excessive radiation exposure. However, if the valves are disassembled for maintenance, the internal surfaces should be examined.

Conclusions: Based on the above evaluation, it is concluded that the Code requirement is impractical. Therefore, it is recommended that relief be granted provided that (a) visual (VT-3) examination of the internal surfaces of the valves is performed whenever the internal surfaces are made accessible due to disassembly and (b) if the valves have not been disassembled, this fact should be reported by the Licensee in the ISI Summary Report at the end of the interval.

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3:1.7 General (No relief requests)

3.2 Class 2 Components

3.2.1 Pressure Vessels (No relief requests)

3.2.2 Piping

3.2.2.1 Request for Relief No. 15, Examination Category C-F-1 and C-F-2, Items C5.10 and C5.50, Class 2 Piping Welds

NOTE: Request for Relief No. 15 was withdrawn by the Licensee in the January 16, 1990 submittal, which states the following:

> "Request for Relief Number 15 is not required by the Code. In the Class 2 allocations tables, the welds concerning this relief are included in the tables and in the program plan tables. Since the relief request is not required to meet the code, this relief request will be deleted. Changes to the allocation tables and in the program plan tables will be made to insure correct accountability."

3.2.2.2 Request for Relief No. 16 (Part 2 of 2), Examination Category C-C, Item C3.20, Class 2 Piping Integrally Welded Attachments

NOTE: Request for Relief No. 16 was withdrawn by the Licensee in the January 16, 1990 submittal, which states the following:

> "Relief Request Number 16 is not required by the Code. The associated program plan tables will be corrected and updated to assure accurate accounting. This activity will be performed to reflect the "non-requirement" code status since the associated thicknesses are < 5/8" for class 1 and < 3/4" for class 2. Relief Request Number 16 is not required and will be deleted."

3.2.3 Pumps (No relief requests)

[illegible]

3.2.4 Valves (No relief requests)

3.2.5 General (No relief requests)

3.3 Class 3 Components (No relief requests)

3.4 Pressure Tests

3.4.1 Class 1 System Pressure Tests (No relief requests)

3.4.2 Class 2 System Pressure Tests

3.4.2.1 Request for Relief No. 7, System Hydrostatic Test of Three Charging Pumps and Discharge Piping to Discharge Isolation Valves in the Chemistry and Volume Control System

Code Requirement: Section XI, Paragraph IWC-5222(a) requires that the system hydrostatic test pressure be at least 1.10 times the system pressure for systems with Design Temperature of 200°F or less, and at least 1.25 times the system pressure for systems with Design Temperature above 200°F. The system pressure shall be the lowest pressure setting among the number of safety or relief valves provided for overpressure protection within the boundary of the system to be tested. For systems (or portions of systems) not provided with safety or relief valves, the system design pressure shall be substituted for the system pressure.

Licensee's Code Relief Request: Relief is requested from performing the Code-required hydrostatic test of Charging Pumps 1, 2, and 3 and associated discharge piping to discharge isolation valves in the Chemistry and Volume Control System at the required pressure of 3420 psig.

Licensee's Proposed Alternative Examination: The Licensee states that, during the hydrostatic test and associated VT-2

[illegible]

visual examination, the charging pumps and associated discharge piping to the first isolation valves will be tested at a pressure of 2400 psig.

Licensee's Basis for Requesting Relief: The Licensee states that the charging pumps have a hydrostatic test pressure limitation on the seals of 2400 psig, as specified by the pump manufacturer. As a result, the pumps and associated discharge piping to the first isolation valves cannot be tested to the required Code test pressure.

Evaluation: The pump design does not permit hydrostatic testing of the CVC charging pumps and associated discharge piping to the first isolation valves at the Code-required test pressure without overpressurizing the pump seals. Therefore, the Code-required hydrostatic test pressure is impractical to attain. The Licensee's proposed alternative test at the reduced pressure of 2400 psig will provide reasonable assurance of the continued inservice structural integrity.

Conclusions: Based on the above, it is concluded that the Code-required hydrostatic test is impractical to perform. Therefore, it is recommended that relief be granted as requested.

3.4.2.2 Request for Relief No. 8, System Hydrostatic Test of Reactor Coolant System Overpressure Protection Nitrogen Accumulator System Valves PCV 430 and PCV 431C

Code Requirement: Section XI, Paragraph IWC-5222(a) requires that the system hydrostatic test pressure be at least 1.10 times the system pressure for systems with Design Temperature of 200°F or less, and at least 1.25 times the system pressure for systems with Design Temperature above 200°F. The system pressure shall be the lowest pressure setting among the number

of safety or relief valves provided for overpressure protection within the boundary of the system to be tested. For systems (or portions of systems) not provided with safety or relief valves, the system design pressure shall be substituted for the system pressure.

Licensee's Code Relief Request: Relief is requested from performing the Code-required hydrostatic test of the reactor coolant system (RCS) overpressure protection nitrogen accumulator system valves PCV 430 and PCV 431C at the required test pressure of 137.5 psig.

Licensee's Proposed Alternative Examination: The Licensee states that the RCS overpressure nitrogen accumulator system will be tested per the Code requirements up to the flex connection to the valve operator. In addition, an inservice pressure test at operating pressure will be performed once each inspection period on the piping, including the diaphragm.

Licensee's Basis for Requesting Relief: The Licensee states that the diaphragms in the subject valve operators are only designed to withstand a maximum pressure of 105 psig and, therefore, cannot be tested to the required Code test pressure.

Evaluation: The system's design does not permit hydrostatic testing of the RCS overpressure protection nitrogen accumulator system valves PCV 430 and PCV 431C at the Code-required test pressure without overpressurizing the diaphragms in the operators of these valves. Therefore, the Code-required hydrostatic test is impractical to perform. The inservice pressure test, which includes the diaphragms, will provide reasonable assurance of the continued inservice structural integrity.

Conclusions: Based on the above, it is concluded that the Code-required hydrostatic test is impractical to perform. Therefore, it is recommended that relief be granted as requested.

3.4.2.3 Request for Relief No. 9, System Hydrostatic Test of the Secondary Side of the Steam Generator and Associated Main Steam Piping

Code Requirement: Section XI, Paragraph IWC-5222(a) requires that the system hydrostatic test pressure be at least 1.10 times the system pressure for systems with Design Temperature of 200°F or less, and at least 1.25 times the system pressure for systems with Design Temperature above 200°F. The system pressure shall be the lowest pressure setting among the number of safety or relief valves provided for overpressure protection within the boundary of the system to be tested. For systems (or portions of systems) not provided with safety or relief valves, the system design pressure shall be substituted for the system pressure.

Licensee's Code Relief Request: Relief is requested from performing the Code-required hydrostatic test of the secondary side of the Steam Generator and associated Main Steam piping at the required pressure of 1356 psig.

Licensee's Proposed Alternative Examination: The Licensee states that the secondary side of the Steam Generator and associated Main Steam piping will be tested at a pressure of 1194 psig, which corresponds to 1.10 times the system pressure.

Licensee's Basis for Requesting Relief: The Licensee states that a pressure differential limitation of 800 psig between the primary and secondary side of the Steam Generator has been adopted. This was established early in plant life due to the

[illegible]

experiences of some plants with primary side tubesheet cladding separation. To maintain this 800 psig differential and the required pressure on the secondary side, the primary system must be heated to a minimum of 160°F which would result in a problem with heat balance and a potential operational problem during implementation of the test procedure. The administrative controls necessary to ensure a proper and safe test and the complexity required for the test procedure result in a situation that should be minimized.

In addition to the Section XI volumetric and surface examination requirements, the piping is part of the augmented inspection program since it falls within the high energy break criteria.

The Licensee's January 16, 1990 submittal states the following:

> "The 800 psi differential pressure limitation was established early in the plant life of R. E. Ginna. This limitation was initiated after several Westinghouse Steam Generators with explosively welded tubesheet cladding experienced primary side tubesheet cladding separation. The hydrostatic test causes the Steam Generator tubesheet to deflect due to reverse loading (in the direction opposite to normal operation). The pressure limitation of 800 psi is an attempt to limit tubesheet deflection and the potential for cladding failure.
>
> To maintain an 800 psi differential pressure across the tubesheet while performing a Code acceptable hydrostatic test of 1356 psi (1.25 x 1085 psi), the required primary system pressure would be 556 psi (1356 psi - 800 psi).
>
> During the period of time that a hydrostatic test and associated examination would be performed, the plant is operating with the Low Temperature Overpressure Protection System (LTOP) set points in effect. The LTOP set points are used to limit primary system pressure to 400 psi or less. In order to pressurize the primary system to 556 psi (a value greater than 400 psi) the LTOP set points must be changed to higher values and additionally the steam generators are required to be functional due to primary system pressure. When the steam generators are functional,

[illegible]

safety valves and level instrumentation are also required to be functional. Safety valves cannot be functional during a hydrostatic test due to gagging requirements and level instrumentation cannot be functional since the steam generators are water solid and therefore level readings would be off scale.

LTOP set points are changed when the primary system temperature reaches 350 degree F, thus there is the potential of a Technical Specification violation if the primary system is pressurized to 556 psi. Technical Specification 3.1.1.2 limits the tubesheet differential temperature to 100 degree F while the potential for a differential temperature of 290 degree F (350 deg. F - 60 deg. F) exists.

In order to alleviate these problems, the Steam Generator and downstream piping, to the class boundary, will be pressurized to 1194 psi (1.1 x 1085 psi) instead of 1356 psi (1.25 x 1085 psi). This action will minimize the potential for tubesheet problems while fulfilling the intent of a hydrostatic test to insure the integrity of the pressure boundary.

In addition to the above test, system pressure tests will be performed each period at system operating pressure and temperature to assure system integrity. The steam generator and associated systems are also subjected to volumetric and surface (Section XI) examinations. The Feedwater and Main Steam piping is also included in an augmented inservice inspection program to provide additional assurances of the system integrity."

Evaluation: The system's design does not permit pressurizing the secondary side of the Steam Generator and associated Main Steam piping to the Code-required test pressure without the potential of cladding failure due to reverse loading. Therefore, the Code-required hydrostatic test is impractical to perform. Pressurizing the components to the alternative test pressures will provide reasonable assurance of the continued inservice structural integrity.

Conclusions: Based on the above, it is concluded that the Code-required hydrostatic test of the secondary side of the Steam Generator and associated Main Steam piping is

impractical. Therefore, it is recommended that relief be granted as requested.

3.4.2.4 Request for Relief No. 13, System Hydrostatic Test of Non-ISI Classified Systems

Code Requirement: Section XI, Paragraph IWC-5222(a) requires that the system hydrostatic test pressure be at least 1.10 times the system pressure for systems with Design Temperature of 200°F or less, and at least 1.25 times the system pressure for systems with Design Temperature above 200°F. The system pressure shall be the lowest pressure setting among the number of safety or relief valves provided for overpressure protection within the boundary of the system to be tested. For systems (or portions of systems) not provided with safety or relief valves, the system design pressure shall be substituted for the system pressure.

Licensee's Code Relief Request: Relief is requested from performing the Code-required hydrostatic test of the non-ISI classified systems, which do not carry radioactive gases or fluids, that contain lines penetrating primary containment.

Relief is requested for the following lines:

Penetration	Description
120	Nitrogen Fill to Safety Injection Accumulators; 1-SI-903
121	Nitrogen Supply to Pressurizer Relief Tank; 3/4-RC-151
121	Reactor Makeup Water to Pressurizer Relief Tank; 2-RC-151
124	Post Accident Test Point Containment Atmosphere
124	Post Accident Test Point Containment Recirculating Filter and Cooling Unit C Discharge
132	Containment Depressurization and Mini-Purge Exhaust
202	Hydrogen Supply to Recombiner B Main Burner
202	Hydrogen Supply to Recombiner B Pilot Burner
203	Post Accident Test Point Containment Atmosphere

(continued)

Penetration	Description
203	Post Accident Test Point Containment Recirculation Filter and Cooling Unit D Discharge
210	Recombiner Oxygen Makeup to Containment
301	Heating Steam Supply to Containment; 2-HS-150-4
303	Heating Steam Condensate Return from Containment
304	Hydrogen Supply to Recombiner A Main Burner
304	Hydrogen Supply to Recombiner A Pilot Burner
305	Post Accident Test Point Containment Atmosphere
305	Post Accident Test Point Containment Recirculating Filter and Cooling Unit A Discharge
305	Post Accident Test Point Containment Recirculating Filter and Cooling Unit B Discharge
305	Containment Air Sample Inlet
305	Containment Air Sample Outlet
307	Fire Service Water
309	Containment Mini-Purge Supply
310	Instrument Air Supply to Containment; 2-IA-125-8
310	Service Air Supply to Containment
313	Containment Leakrate Test Connection Depressurization
317	Containment Leakrate Test Connection Supply; 6-AT-150-4
324	Demineralized Water
332	Hydrogen Monitor A Inlet
332	Hydrogen Monitor B Inlet
332	Hydrogen Monitor Outlet

Licensee's Proposed Alternative Examination: The Licensee proposes to test these lines in accordance with 10 CFR 50 Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactor," commensurate with the safety function the line performs in accordance with Technical Specification, Surveillance requirements. Additionally, at least once each period, exposed portions of the lines penetrating primary containment will be examined during normal system operation.

Licensee's Basis for Requesting Relief: The Licensee states that the safety function of these lines is to become part of the containment isolation system during periods when containment isolation is required. Therefore, the pressure testing requirements should be based on the containment system design not the associated process system design requirements.

[illegible]

The Licensee's January 16, 1990 submittal states the following:

> "Due to the vintage of the R. E. Ginna Plant, the piping within these penetration boundaries was not constructed to Class 2 requirements. Regulatory Guide 1.26 does not require these penetrations to be classified as Quality Group B or Class 2. However, Rochester Gas and Electric has optionally classified these lines Class 2. For optionally classified piping, IWA-1320(e) allows the Owner the option to apply the Section XI rules. In the case of these lines, Rochester Gas & Electric has concluded that Appendix J testing and a VT-2 examination once per period during normal system operation will provide testing commensurate with the safety function. In addition, these lines are now subject to ASME Section XI rules for Repairs and Replacements."

Evaluation: Although these lines were not constructed to Class 2 requirements, the Licensee has not shown that the system's safety criteria permit these lines to be nonnuclear safety class. Therefore, paragraph IWA-1320(e) may not be applicable.

In addition, Appendix J addresses the structural integrity of the primary containment, not the structural integrity of the Code Class 2 process piping for which relief is being requested. The request for relief does not contain information about the pressures involved. The proposed Appendix J test pressure may be much lower than the Code-required test pressure, in which case the inservice structural integrity would not be identified. Also, the proposed Appendix J test does not identify the location of a leak if present.

Conclusions: Based on the above evaluation, it is concluded that the Licensee has not justified the determination of impracticality. Therefore, it is recommended that relief be denied.

[illegible]

3.4.3 Class 3 System Pressure Tests

3.4.3.1 Request for Relief No. 6, System Hydrostatic Test of the Radioactive Waste Holdup Tank

NOTE: Request for Relief No. 6 was withdrawn by the Licensee in the January 16, 1990 submittal, which states the following:

> "This request is specific to the Code hydrostatic test requirements concerning the waste hold-up tank. Liquid waste from various locations in the plant including the reactor coolant drain tank, chemical drain tank, intermediate and auxiliary building sumps, floor, laundry and shower drains are diverted to the waste hold-up tank. During cold shut down and refueling outages, the level in this tank varies depending on the activities performed during an outage. If required to perform the Code examination when the tank is full, there is a possibility of a hazardous situation occurring. Liquid waste could be diverted to the waste hold-up tank while performing the required Code static head examination. This action could result in a liquid radioactive waste spill at one of various locations.
>
> It was the intent of this relief request to prevent a hazardous event from occurring. Upon review of your inquiry, we have concluded that it is possible to perform the required examination on this component. It will require greater coordination with Operations and plant outage activities than is normally expected, but it can be performed. Therefore, Relief Request Number 6 is no longer required."

3.4.3.2 Request for Relief No. 10, System Hydrostatic Test of the Standby Auxiliary Feedwater Pump Recirculation Line

Code Requirement: Section XI, Paragraph IWD-5223(a) requires that the system hydrostatic test pressure be at least 1.10 times the system pressure for systems with Design Temperature of 200°F or less, and at least 1.25 times the system pressure for systems with Design Temperature above 200°F. The system pressure shall be the lowest pressure setting among the number of safety or relief valves provided for overpressure protection within the boundary of the system to be tested. For systems

(or portions of systems) not provided with safety or relief valves, the system design pressure shall be substituted for the system pressure.

Licensee's Code Relief Request: Relief is requested from performing the Code-required hydrostatic test of the Standby Auxiliary Feedwater Pump recirculation line between valves AOV 9710A and AOV 9710B and their associated downstream flow orifices.

Licensee's Proposed Alternative Examination: The Licensee states that the Class 3 portion of this piping shall be VT-2 examined at operational discharge pressure once each period during functional testing.

Licensee's Basis for Requesting Relief: The Licensee states that, in order to hydrostatically test this piping to Section XI requirements, the pressure reducing flow orifices downstream of valve AOV 9710A and B would require removal and blank flanges installed. System piping does not provide an isolation valve between the orifices and the Condensate Supply Tank. A significant reduction in tank level would be required to facilitate orifice removal, which is considered to be impractical.

Evaluation: The Licensee has not provided adequate information to justify the determination of impracticality. Information that should be addressed includes (a) why the Licensee considers it impractical to lower the Condensate Supply Tank level and (b) whether or not the use of freeze plugs has been considered and, if so, why it would be impractical to use them.

Conclusions: Based on the above, it is concluded that the Licensee has not demonstrated that the Code-required hydrostatic test is impractical. Therefore, it is recommended that relief be denied.

3.4.3.3 <u>Request for Relief No. 11, System Hydrostatic Test of the Boric Acid Filter and Associated Piping in the Chemical and Volume Control System</u>

NOTE: Request for Relief No. 11 was withdrawn by the Licensee in the January 16, 1990 submittal, which states the following:

> "The flange gasket was reviewed with new technical information that eliminated concerns over the safe working pressure of the Boric Acid Filter housing flange gasket.
>
> Upon review of your inquiry, it has been concluded that the Hydrostatic test is possible and will be incorporated into the plan. Therefore, Relief Request Number 11 is no longer required."

3.4.3.4 <u>Request for Relief No. 12, System Hydrostatic Test of Portions of the Emergency Diesel Generation System</u>

<u>Code Requirement</u>: Section XI, Paragraph IWD-5223(a) requires that the system hydrostatic test pressure be at least 1.10 times the system pressure for systems with Design Temperature of 200°F or less, and at least 1.25 times the system pressure for systems with Design Temperature above 200°F. The system pressure shall be the lowest pressure setting among the number of safety or relief valves provided for overpressure protection within the boundary of the system to be tested. For systems (or portions of systems) not provided with safety or relief valves, the system design pressure shall be substituted for the system pressure.

<u>Licensee's Code Relief Request</u>: Relief is requested from performing the Code-required hydrostatic test of the following portions of the Emergency Diesel Generation System:

(a) Starting Air, including receiver tanks and associated piping,

[illegible]

(b) Fuel Oil Transfer pumps, suction and discharge lines, including miscellaneous lines terminating at oil storage tanks, and

(c) Jacket Cooling Water system, including miscellaneous lines terminating at the cooling water expansion tanks.

Licensee's Proposed Alternative Examination: The Licensee states that inservice testing shall be performed on the Air Start System at least once each period in accordance with the requirements of Section XI. Additionally, once each quarter, a pressure decay test shall be performed on the air receiver to verify check valve operability in the reverse direction for the air receiver inlet check.

System functional testing shall be performed at least once each period on the Diesel Fuel Oil Transfer and Jacket Cooling Water Systems in accordance with the requirements of Section XI.

In addition to the testing discussed above, Technical Specification 6.4.1 requires surveillance testing to be performed on a monthly basis, such as verifying operability of the fuel oil transfer pumps and that the diesel starts from normal standby conditions.

Licensee's Basis for Requesting Relief: The Licensee states that only portions of the piping associated with the components identified above are capable of being pressure tested.

The Air Start System pressure test would require termination prior to reaching the engine skid to preclude administrating air to the Air Start Motors. This would leave that portion of piping between the Air Start Motors to the first isolation, prior to reaching the engine skid, untestable.

[illegible]

The Diesel Fuel Oil Transfer system would require flange connection disassembly and the installation of blind flanges to isolate the Diesel Oil Storage Tank, which is vented to atmosphere, and the Day Tank where no means is provided to isolate the transfer pump discharge piping at a point close to the day tank. Additionally, the overflow piping from the day tank to the storage tank, which is identified as Class 3, has no isolation valves installed and is vented to the atmosphere.

Due to vents to the atmosphere, the Jacket Cooling Water System would require isolating the Cooling Water Expansion Tank, including most of the piping subject to pressure testing. Due to the amount of piping within the Class boundary which is unable to be pressurized, testing in accordance with Section XI requirements would not prove system integrity over and above the existing surveillance Inservice and Functional Testing.

Evaluation: The system's design does not permit pressurizing all of these Class 3 lines to the Code-required test pressure. Therefore, the Code-required hydrostatic test pressure for the unisolatable portions of this piping is impractical to attain. The proposed alternative testing will provide reasonable assurance of the continued inservice structural integrity.

Conclusions: Based on the above, it is concluded that the Code-required hydrostatic test is impractical. Therefore, it is recommended that relief be granted as requested.

3.4.3.5 Request for Relief No. 14, System Hydrostatic Test of All Pressure Retaining Components Within the Class 3 Portion of the Service Water System

Code Requirement: Section XI, Paragraph IWD-5223(a) requires that the system hydrostatic test pressure be at least 1.10 times the system pressure for systems with Design Temperature

of 200°F or less, and at least 1.25 times the system pressure for systems with Design Temperature above 200°F. The system pressure shall be the lowest pressure setting among the number of safety or relief valves provided for overpressure protection within the boundary of the system to be tested. For systems (or portions of systems) not provided with safety or relief valves, the system design pressure shall be substituted for the system pressure.

Licensee's Code Relief Request: Relief is requested from performing the Code-required hydrostatic test of all pressure retaining components within the Class 3 portion of the Service Water System.

Licensee's Proposed Alternative Examination: The Licensee states that pressure retaining components within the operational boundary will receive an inservice test at operating pressure and an associated VT-2 visual examination each period during the interval.

Licensee's Basis for Requesting Relief: The Licensee states that the hydrostatic test requirement for the Service Water system is impractical due to system design which dictates the use of an open-ended test. The portion of the system downstream of the heat exchanger is also open-ended and cannot be hydrostatically tested. The remaining section of the system is only isolatable by means of butterfly valves which were not designed to provide a leak tight boundary. With the system as such, it would be impractical to expect that the leakages, other than at the valves, could be detected.

The ample margin in cooling capacity inherently provided by system design does not dictate the need for an essentially leak tight boundary. Since the system is in constant operation, its integrity is continually monitored. Thorough inspection of the system each period at the full operating pressure is adequate

to detect any gross failures in the system without degrading system safety or availability.

Evaluation: The Code-required hydrostatic pressure test of isolated portions of Class 3 piping associated with the Service Water System is impractical to perform because the butterfly valves of this system are unable to sustain the pressure required for the hydrostatic test. With regard to the open-ended portions of these lines, paragraph IWD-5223(d) of Section XI should apply. Paragraph IWD-5223(d) states that, for open-ended portions of discharge lines beyond the last shutoff valve in nonclosed systems (e.g., service water systems), confirmation of adequate flow during system operation shall be acceptable in lieu of system hydrostatic test. The inservice test and the confirmation of adequate flow during system operation will provide reasonable assurance of the continued inservice structural integrity.

Conclusions: Based on the above, it is concluded that the required hydrostatic test of these portions of piping is impractical to perform. Therefore, it is recommended that relief be granted as requested.

3.4.4 General (No relief requests)

3.5 General

3.5.1 Ultrasonic Examination Techniques (No relief requests)

3.5.2 Exempted Components (No relief requests)

3.5.3 Other

3.5.3.1 Request for Relief No. 3, Use of an Authorized Inspection Agency to Provide Inspection Services

Code Requirement: Section XI, Paragraph IWA-1400(f) requires an arrangement with an Authorized Inspection Agency to provide inspection services. In addition, the Code requires that certain administrative functions be performed by the "Enforcement Authority" and "Authorized Nuclear Inservice Inspector."

Licensee's Code Relief Request: Relief is requested from using an "Authorized Inspection Agency" for ISI.

Licensee's Proposed Alternative Examination: Records and reports of the inservice inspection will be developed and maintained by Rochester Gas and Electric and include such items as examination plans and schedules, examination results, and corrective actions.

The functions of the authorized nuclear inservice inspector, namely their review and verification of inservice examinations, personnel qualification, and equipment certification during the annual outages at Ginna Nuclear Power Station will be performed by personnel of the Hartford Steam Boiler Inspection and Insurance Company. The qualifications of the inspectors, inspections specialists, and inspection agency are in compliance with the Code.

Licensee's Basis for Requesting Relief: Ginna Nuclear Power Station is located in the State of New York. The Licensee states that this state has not endorsed ASME Codes and, therefore, does not provide administrative organization and controls such as "Enforcement Authority," "Authorized Nuclear

[illegible]

Inservice Inspector," and "Reporting Systems." However, Ginna Station's Quality Assurance Program does provide equivalent administrative control. Therefore, the Licensee requests that Ginna's Station Quality Assurance Program be used in lieu of Code administrative functions.

Rochester Gas and Electric's program for the inservice inspection, governed by the R. E. Ginna Nuclear Power Station Quality Assurance Manual, contains the requirements and responsibilities for implementation of the program and procedures. The procedures have been prepared and approved by the responsible organizations within Rochester Gas and Electric (e.g., Ginna Station, Engineering, Materials Engineering and inspection Services, Electric Meter and Laboratory and Purchasing).

Approved procedures will be implemented to control the standards for examination evaluation. These procedures include identifications of the organization performing the inspection, description of the method of inspection to be used, acceptance and rejection criteria, and requirements for providing evidence of completion and certification of the inspection activity.

In addition, procedures are developed by Ginna Nuclear Power Station to prescribe the disposition of nonconformances. The procedures implemented for the repairs, the retest procedures, and the test results will be reviewed by the Plant Operating Review Committee. Members of this committee include technically qualified staff personnel.

Examination techniques have been established in accordance with written requirements and incorporated into written procedures. Qualifications for nondestructive test personnel are in compliance with Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel."

Evaluation: Although the State of New York does not endorse the ASME Code and, therefore, does not provide administrative organization and controls such as "Enforcement Authority," Authorized Nuclear Inservice Inspector," and "Reporting Systems," equivalent administrative control is provided by the Ginna Nuclear Power Station Quality Assurance Program. Personnel of the Hartford Steam Boiler Inspection and Insurance Company will perform the functions of the Authorized Nuclear Inservice Inspector during the annual outages at Ginna Nuclear Power Station. The Hartford Steam Boiler Inspection and Insurance Company has been designated the Authorized Nuclear Inservice Inspector by other states for other nuclear reactors. Therefore, the intent of the Code will be met by using the Ginna Nuclear Power Station Quality Assurance Program in lieu of the Code administrative functions.

Conclusions: Based on the above, it is concluded that the intent of the Code will be met. Therefore, it is recommended that relief be granted as requested.

[illegible]

4. CONCLUSION

Pursuant to 10 CFR 50.55a(g)(6), it has been determined that certain Section XI required inservice examinations are impractical to perform. In all cases for which relief is requested, except Request for Relief No. 13, the Licensee has demonstrated that specific Section XI requirements are impractical. For Request for Relief Nos. 10 and 13, it is concluded that: (a) the Licensee has not provided information to support the determination that the Code requirement is impractical and (b) requiring the Licensee to comply with the Code requirement would not result in hardship. Request for Relief Nos. 6, 11, 15, and 16 were withdrawn by the Licensee and deleted from the ISI Program Plan.

This technical evaluation has not identified any practical method by which the Licensee can meet all the specific inservice inspection requirements of Section XI of the ASME Code for the existing R. E. Ginna Nuclear Power Station facility. Requiring compliance with all the exact Section XI required inspections would require redesign of a significant number of plant systems, sufficient replacement components to be obtained, installation of the new components, and a baseline examination of these components. Even after the redesign efforts, complete compliance with the Section XI examination requirements probably could not be achieved. Pursuant to 10 CFR 50.55a(g)(6), relief is allowed from these requirements which are impractical to implement if granting the relief will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Therefore, it is concluded that the public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical.

The development of new or improved examination techniques should continue to be monitored. As improvements in these areas are achieved, the Licensee should incorporate these techniques in the ISI program plan examination requirements.

Based on review of the R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection Program Plan, the Licensee's response to the NRC request for additional information, and the recommendations for granting relief from the ISI examination requirements that have been determined to be impractical, it is concluded that the R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection Program Plan, with the exception of Request for Relief Nos. 10 and 13, is acceptable and in compliance with 10 CFR 50.55a(g)(4).

5. REFERENCES

1. Code of Federal Regulations, Volume 10, Part 50.

2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Division 1, 1986 Edition.

3. R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection Program Plan, dated July 21, 1989.

4. NUREG-0800, Standard Review Plans, Section 5.2.4, "Reactor Coolant Boundary Inservice Inspection and Testing," and Section 6.6, "Inservice Inspection of Class 2 and 3 Components," July 1981.

5. Letter, dated November 16, 1989, A. R. Johnson (NRC) to R. C. Mecredy [Rochester Gas and Electric Corporation (RGE)], request for additional information.

6. Letter, dated January 16, 1990, R. C. Mecredy (RGE) to A. R. Johnson (NRC), response to the NRC request for additional information.

7. NRC Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1, August 1975.

8. NRC Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," Revision 1, February 1983.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report presents the results of the evaluation of the R. E. Ginna Nuclear Power Station Third 10-Year Interval Inservice Inspection (ISI) Program Plan, submitted July 21, 1989, including the requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements which the Licensee has determined to be impractical. The R. E. Ginna Nuclear Power Station Third 10-Year Interval ISI Program Plan is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during previous Nuclear Regulatory Commission (NRC) reviews. The requests for relief are evaluated in Section 3 of this report.

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