September 28, 2000

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

SUBJECT: REQUESTS FOR RELIEF NOS. R-42 (REVISION 1), R-43, R-1, R-2, R-3, R-4, R-5, R-6, R-7, R-8, R-14, AND R-15 FROM THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE (ASME CODE) REQUIREMENTS FOR THE R. E. GINNA NUCLEAR POWER PLANT THIRD AND FOURTH 10-YEAR INTERNAL INSPECTION INTERVALS (TAC NOS. MA6821, MA6822, AND MA8058)

Dear Dr. Mecredy:

By letter dated October 5, 1999, as supplemented by letter dated May 2, 2000, you requested Relief Request No. RR-42, and Relief Request No. RR-43 from certain ASME Code Section XI inservice examination requirements for the third 10-year inservice inspection (ISI) interval. By letter dated October 8, 1999, as supplemented by letter dated May 2, 2000, you requested Relief Requests Nos. RR-1 though RR-15 from ASME Code Section XI inservice examination requirements for the fourth 10-year inservice inspection interval. By letter dated March 21, 2000, the Relief Requests RR-9 through RR-13 from the Code requirements were granted for the fourth 10-year interval.

The NRC staff concludes that for the third 10-year ISI interval Requests for Relief RR-42, Rev. 1 (Parts B and C), and RR-43 the Code requirements are impractical and the proposed alternatives provide reasonable assurance of structural integrity of the subject components discussed in your requests for relief. Therefore, relief is granted for Requests for Relief RR-42, Rev. 1 (Parts B and C) and RR-43, pursuant to 10 CFR 50.55a(g)(6)(i). The staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and 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.

For Relief RR-42, Rev. 1 (Part A) the staff concludes the imposition of the Code requirements would result in a significant hardship without a compensating increase in the level of quality and safety. The staff concludes that your alternative provides reasonable assurance of structural integrity of the subject components discussed in your requests for relief. Therefore, your alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year interval.

The staff concludes that, for the fourth 10-year ISI interval Requests for Relief RR-3, RR-6, RR-7, RR-8, RR-14 (Parts A and B), and RR-15, the Code requirements are impractical and the proposed alternatives provide reasonable assurance of structural integrity of the subject components discussed in your requests for relief. Therefore, relief is granted for Requests for Relief RR-3, RR-6, RR-7, RR-8, RR-14 (Parts A and B), and RR-15 pursuant to

10 CFR 50.55a(g)(6)(i). The staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and 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.

The alternatives contained in Requests for Relief RR-1, Rev. 1, and RR-5 provide an acceptable level of quality and safety and are authorized pursuant to 10 CFR 50.55a(a)(3)(i). For Request for Relief RR-1 the alternative is authorized for the fourth 10-year interval. For Request for Relief RR-5 the use of Code Cases 2142 and 2143 are authorized for the fourth 10-year interval or until such time Code Cases 2142 and 2143 are referenced in a future revision of 10 CFR 50.55a. At that time, if you intend to continue to implement Code Cases 2142 and 2143, you should follow all provisions in the subject code cases with limitations (if any) referenced in 10 CFR 50.55a.

For RR-4, the imposition of the Code requirements would result in a significant hardship without a compensating increase in the level of quality and safety. The staff concludes that for Request for Relief RR-4, the alternatives provide reasonable assurance of structural integrity of the subject components. Therefore, your alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year interval.

For Request for Relief RR-2, the staff determined that based upon your proposed alternative the Code requirements are being met. Therefore, relief is not required.

The staff's evaluations and conclusions are contained in the enclosed safety evaluation.

Sincerely,

/RA/

Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure: Safety Evaluation
w/attachment

cc w/encl: See next page

10 CFR 50.55a(g)(6)(i). The staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and 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.

The alternatives contained in Requests for Relief RR-1, Rev. 1, and RR-5 provide an acceptable level of quality and safety and are authorized pursuant to 10 CFR 50.55a(a)(3)(i). For Request for Relief RR-1 the alternative is authorized for the fourth 10-year interval. For Request for Relief RR-5 the use of Code Cases 2142 and 2143 are authorized for the fourth 10-year interval or until such time Code Cases 2142 and 2143 are referenced in a future revision of 10 CFR 50.55a. At that time, if you intend to continue to implement Code Cases 2142 and 2143, you should follow all provisions in the subject code cases with limitations (if any) referenced in 10 CFR 50.55a.

For RR-4, the imposition of the Code requirements would result in a significant hardship without a compensating increase in the level of quality and safety. The staff concludes that for Request for Relief RR-4, the alternatives provide reasonable assurance of structural integrity of the subject components. Therefore, your alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year interval.

For Request for Relief RR-2, the staff determined that based upon your proposed alternative the Code requirements are being met. Therefore, relief is not required.

The staff's evaluations and conclusions are contained in the enclosed safety evaluation.

Sincerely,
/RA/
Marsha Gamberoni, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure: Safety Evaluation
w/attachment
cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THIRD AND FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION

REQUESTS FOR RELIEF FOR

R.E. GINNA NUCLEAR POWER PLANT

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NUMBER 50-244

1.0 INTRODUCTION

Inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code) and applicable addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The Code of record for the Ginna Nuclear Power Plant third 10-year ISI interval is the 1986 Edition of the ASME Code and the Code of record for the fourth 10-year ISI interval is the 1995 Edition through 1996 Addenda of the ASME Code.

2.0 EVALUATION

The NRC staff, with technical assistance from Idaho National Engineering and Environmental Laboratory (INEEL), has reviewed the information concerning inservice inspection (ISI) program requests for relief submitted for the third and forth 10-year intervals for R.E. Ginna Nuclear

Power Plant by Rochester Gas and Electric Corporation (the licensee) in two separate letters dated October 5, 1999, and one letter dated October 8, 1999. Additional information was provided in the licensee's two separate letters dated May 2, 2000.

For the R.E. Ginna Nuclear Power Plant relief is granted from the inspection requirements which have been determined to be impractical to perform. Alternatives are authorized where it provides an acceptable level of quality and safety, or where compliance would result in a hardship or unusual difficulty without a compensating increase in quality or safety.

3.0 CONCLUSION

The R.E. Ginna Nuclear Power Plant requests for relief from and alternatives to the Code requirements have been reviewed by the staff with the assistance of its contractor, INEEL. The attached Technical Letter Report (TLR), prepared by INEEL provides the contractor's evaluation of these relief requests and alternatives. The staff has reviewed the TLR and adopts the evaluations and recommendations for granting relief or authorizing alternatives. The TLR lists each relief request and the status of approval.

The staff concludes that for the third 10-year ISI interval Requests for Relief RR-42, Rev. 1 (Parts B and C), and RR-43, as evaluated by this Safety Evaluation, the Code requirements are impractical and the proposed alternatives provide reasonable assurance of structural integrity of the subject components discussed in the licensee's requests for relief. Therefore, relief is granted for Requests for Relief RR-42, Rev. 1 (Parts B and C) and RR-43, pursuant to 10 CFR 50.55a(g)(6)(i). The staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and 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.

For Relief RR-42, Rev. 1 (Part A) the imposition of the Code requirements would result in a significant hardship without a compensating increase in the level of quality and safety. The staff concludes that the licensee's alternative provides reasonable assurance of structural integrity of the subject components discussed in the licensee's requests for relief. Therefore the licensee's alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year interval.

The staff concludes that for the fourth 10-year ISI interval Requests for Relief RR-3, RR-6, RR-7, RR-8, RR-14 (Parts A and B), and RR-15 as evaluated by this Safety Evaluation, the Code requirements are impractical and the proposed alternatives provide reasonable assurance of structural integrity of the subject components discussed in the licensee's requests for relief. Therefore, relief is granted for Requests for Relief RR-3, RR-6, RR-7, RR-8, RR-14 (Parts A and B), and RR-15 pursuant to 10 CFR 50.55a(g)(6)(i). The staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and 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.

The alternatives contained in Requests for Relief RR-1, Rev. 1, and RR-5 provide an acceptable level of quality and safety and are authorized pursuant to 10 CFR 50.55a(a)(3)(i). For Request for Relief RR-1 the alternative is authorized for the fourth 10-year interval. For Request for Relief RR-5 the use of Code Cases 2142 and 2143 are authorized for the fourth 10-year interval or until such time Code Cases 2142 and 2143 are referenced in a future revision of 10 CFR 50.55a. At that time, if the licensee intends to continue to implement Code Cases 2142 and 2143, the licensee should follow all provisions in the subject code cases with limitations (if any) referenced in 10 CFR 50.55a.

For RR-4, the imposition of the Code requirements would result in a significant hardship without a compensating increase in the level of quality and safety. The staff concludes that for Request for Relief RR-4, the alternatives provide reasonable assurance of structural integrity of the subject components. Therefore, the licensee's alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year interval.

For Request for Relief RR-2, the staff determined that based upon the licensee's proposed alternative the Code requirements are being met. Therefore, relief is not required.

Principal Contributor: T. McLellan

Date: September 28, 2000

TECHNICAL LETTER REPORT
THIRD AND FOURTH 10-YEAR INTERVALS INSERVICE INSPECTION
REQUESTS FOR RELIEF
FOR
ROCHESTER GAS AND ELECTRIC CORPORATION
ROBERT EMMIT GINNA NUCLEAR POWER PLANT
DOCKET NUMBER: 50-244

1. INTRODUCTION

By two separate letters dated October 5, 1999, and one letter dated October 8, 1999, the licensee, Rochester Gas and Electric Corporation, submitted requests for relief from the requirements of the ASME Code, Section XI, for the Robert Emmit Ginna Nuclear Power Plant (Ginna) third and fourth 10-year inservice inspection (ISI) intervals. In response to an NRC Request for Additional Information (RAI), the licensee provided additional information in two separate letters dated May 2, 2000. Clarifying information was provided by the licensee in electronic mail dated May 17, 2000. The Idaho National Engineering and Environmental Laboratory (INEEL) staff's evaluation of the subject requests for relief is in the following section.

2. EVALUATION

The information provided by Rochester Gas and Electric Corporation in support of the requests for relief from Code requirements has been evaluated and the bases for disposition are documented below. The Code of record for the Ginna Nuclear Power Plant third 10-year ISI interval, which began January 1, 1990, is the 1986 Edition of Section XI of the ASME Boiler and Pressure Vessel Code. The Code of record for the fourth 10-year ISI interval, which began January 1, 2000, is the 1995 Edition through 1996 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code.

2.1 Proposed Alternative to 10 CFR 50.55a(g)(6)(ii)(A), Request for Relief No. 42 Revision 1 (Part A), Augmented Reactor Pressure Vessel Examination (Third Ten-Year Interval)

Regulatory Requirement: In accordance with 10 CFR 50.55a(g)(6)(ii)(A), all licensees must implement once, as part of the inservice inspection interval in effect on September 8, 1992, an augmented examination of the reactor pressure vessel (RPV) welds specified in Item B1.10 of Examination Category B-A of the 1989 Edition of the ASME Code, Section XI. Examination Category B-A, Items B1.11 and B1.12 require volumetric examination of essentially 100% of the RPV circumferential and longitudinal shell welds, as defined by Figures IWB-2500-1 and -2, respectively. Essentially 100%, as defined by 10 CFR 50.55a(g)(6)(ii)(A)(2), is greater than 90% of the examination volume of each weld.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(g)(6)(ii)(A)(5), the licensee proposed that the coverage obtained be found acceptable for the following weld.

Comp. ID	Component Description	ASME Category	Item Number	Aggregate Coverage	Limitation
RPV-D	Lower Shell-to-Ring Forging Circumferential Weld	B-A	B1.11	81%	Guide Lugs and Incore Instrumentation

The licensee stated:

"R.E. Ginna Nuclear Power Plant proposes that the volumetric examination coverage identified within Attachment Number 1[1] be acceptable in fulfilling required volumetric examination coverage."

Licensee's Basis for Proposed Alternative (as stated:

"The Reactor Pressure Vessel (RPV) was designed and constructed to ASME Section III, 1965 Edition. This code did not contain requirements to ensure that items be accessible for future examinations. The eight (8) items identified within Attachment Number 1 were installed utilizing this construction code which did not provide for accessibility for future ISI NDE.

"The Class 1 Reactor Pressure Vessel is part of the ASME Section XI VT-2 Leakage Examination boundary. Class 1 Leakage Examinations are performed each refueling outage as required by the Code to insure pressure boundary integrity. In addition to the ASME Section XI leakage examinations, a very sensitive leak detection system, governed by Technical Specification 3.4.15, can detect minimal RCS leakage. The combination of RCS leak detection capability and the Class 1 leakage examination that is performed each refueling outage provide additional assurances in maintaining plant safety.

"All possible means were investigated to increase examination coverage. Radiographic examination is impractical due to the vessel being full of water and high background radiation would mask or prevent meaningful results. Ultrasonic technique variations were employed to maximize examination coverage utilizing latest PDI technology using both 70 degree and 45 degree shear and longitudinal waves from the vessel inside surface. The vessel OD surface was also investigated but this surface is enclosed by a permanent rivetted (sic) sheathing and asbestos insulation that has never been removed. To access welds from the outside vessel surface would require scaffolding, building enclosures, removal of the rivetted sheathing, cleaning, disposal, reinstallation of the sheathing and scaffold removal. This major activity would expose work personnel to a high radiation dose of 150 to 200 Rem. There is no current industry safety concerns associated with these welds and previous examination history show no active degraded conditions exist. No additional coverage

1. Attachments, sketches and drawings are not included in this report.

beyond the identified coverage in Attachment Number 1 can be achieved that is practical and safe for the work force.

"ASME Section XI Class 1 periodic leakage examinations are performed, and sensitive RCS leak detection capability also exists. This leak detection system and periodic system leakage examinations provide additional assurances in maintaining plant safety. The identified volumetric examination coverage for these items should be acceptable in fulfilling coverage requirements."

Evaluation: To comply with the augmented reactor vessel examination requirements of 10 CFR 50.55a(g)(6)(ii)(A), licensees must volumetrically examine essentially 100% of each of the Item B1.10 shell welds. As an alternative to the requirement of the Regulations, the licensee proposed that the examination coverage obtained be considered to provide an acceptable level of quality and safety for the subject RPV weld.

At Ginna Nuclear Power Plant, Unit 1, the augmented coverage requirements could not be met for lower shell-to-ring forging circumferential Weld RPV-D, due to core guide lugs and in-core instrumentation. For Weld RPV-D, these physical obstructions limited coverage to 81% of the required volume; to achieve complete coverage, design modifications would be required to increase access from the inside surface (ID).

As a result of the augmented volumetric examination rule, licensees must make a reasonable effort to maximize examination coverage of their reactor vessel welds. In cases where examination coverage from the ID is inadequate, examination from the outside surface (OD) using manual inspection techniques may be an option. However, extensive surface preparation (removal of insulation, weld preparation and scaffolding erection) would be required for the licensee to perform supplemental examinations from the outside diameter. The effort expended to obtain access to the examination surface from the OD would result in considerable radiological exposure that is not warranted for the additional volumetric coverage that may be achieved. Therefore, imposition of this requirement would result in a significant hardship without a compensating increase in the level of quality and safety.

The licensee has examined a significant portion of the subject weld (81%). In addition, the licensee has met the coverage requirements for the remaining Item B1.10 RPV shell welds. Based on the volumetric examination coverage attained, and the examinations conducted on other B1.10 RPV welds, the INEEL staff concludes that any significant patterns of degradation, if present, would have been detected and that the examinations performed provide reasonable assurance of the continued structural integrity of the subject welds. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

2.2 Request for Relief No. 42 Revision 1 (Part B), Examination Category B-A, Items B1.11, and B1.30, Pressure-Retaining Welds in Reactor Vessels (Third Ten-Year Interval)

Code Requirement: Examination Category B-A, Items B1.11 and B1.30 require 100% volumetric examination of the Reactor Pressure Vessel Circumferential Shell Welds and

Shell-to-Flange Welds, as defined by Figures IWB-2500-1 and IWB-2500-4, respectively.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% volumetric examination of the welds listed below.

Comp. ID	Component Description	ASME Category	Item Number	Coverage	Limitation
RPV-A	Vessel to Flange Weld	B-A	B1.30	54%	Obstructed by keyways and irradiation slots
RPV-D	Lower shell to ring forging	B-A	B1.11	81%	Obstructed by guide lugs and incore instrumentation

Licensee's Basis for Requesting Relief (as stated):

"Relief/Exemption is requested pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii)..., in that the required examination coverage for the identified items is impractical and would require reactor vessel component redesign or replacement to enable the required inspection coverage.

"The Reactor Pressure Vessel (RPV was designed and constructed to ASME Section III, 1965 Edition. This code did not contain requirements to ensure that items be accessible for future examinations. The eight (8) items identified within Attachment Number 1 were installed utilizing this construction code which did not provide for accessibility for future ISI NDE.

"The Class 1 Reactor Pressure Vessel is part of the ASME Section XI VT-2 Leakage Examination boundary. Class 1 Leakage Examinations are performed each refueling outage as required by the Code to insure pressure boundary integrity. In addition to the ASME Section XI leakage examinations, a very sensitive leak detection system, governed by Technical Specification 3.4.15, can detect minimal RCS leakage. The combination of RCS leak detection capability and the Class 1 leakage examination that is performed each refueling outage provide additional assurances in maintaining plant safety.

"All possible means were investigated to increase examination coverage. Radiographic examination is impractical due to the vessel being full of water and high background radiation would mask or prevent meaningful results. Ultrasonic technique variations were employed to maximize examination coverage utilizing latest PDI technology using both 70 degree and 45 degree shear and longitudinal waves from the vessel inside surface. The vessel OD surface was also investigated but this surface is enclosed by a permanent rivetted sheathing and asbestos insulation that has never been removed. To access welds from the outside vessel surface would require scaffolding, building enclosures, removal of the rivetted sheathing, cleaning, disposal, reinstallation of the sheathing and scaffold removal. This major activity would expose work

personnel to a high radiation dose of 150 to 200 Rem. There is no current industry safety concerns associated with these welds and previous examination history show no active degraded conditions exist. No additional coverage beyond the identified coverage in Attachment Number 1 can be achieved that is practical and safe for the work force.

"ASME Section XI Class 1 periodic leakage examinations are performed, and sensitive RCS leak detection capability also exists. This leak detection system and periodic system leakage examinations provide additional assurances in maintaining plant safety. The identified volumetric examination coverage for these items should be acceptable in fulfilling coverage requirements."

Licensee's Proposed Alternative Examination (as stated):

"R.E. Ginna Nuclear Power Plant proposes that the volumetric examination coverage identified within Attachment Number 1 be acceptable in fulfilling required volumetric examination coverage."

Evaluation: The Code requires that the subject reactor pressure vessel welds be 100% volumetrically examined during the inspection interval. Due to access limitations resulting from keyways, irradiation sample slots, core guide lugs and in-core instrumentation, examinations were limited to 54% and 81% of the required volume of the subject welds. Based on the information provided in this request for relief, it is concluded that examination of the subject welds to the extent required by the Code is impractical. For complete examination, redesign and modification of the reactor vessel would be necessary. Imposition of this requirement would cause a considerable burden on the licensee.

Based upon the volumetric examination of the accessible portion of the subject welds, and the volumetric examinations of other RPV welds, it is concluded that significant patterns of degradation, if present, would have been detected. As a result, reasonable assurance of continued structural integrity has been provided. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted.

2.3 Request for Relief No. 42 Revision 1(Part C), Examination Category B-D, Full Penetration Welds of Nozzles in Vessels (Third Ten-Year Interval)

Code Requirement: Examination Category B-D, Items B3.90 and B3.100, require 100% volumetric examination of nozzle-to-vessel welds and nozzle inside radius sections, as defined by Figure IWB-2500-7.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% volumetric examination of the reactor vessel nozzle welds listed below.

Comp. ID	Component Description	ASME Category	Item Number	Coverage	Limitation
N1A	Nozzle to Vessel	B-D	B3.90	55%	Configuration of Nozzle Boss
N1B	Nozzle to Vessel	B-D	B3.90	55%	Configuration of Nozzle Boss
AC-1002	Nozzle to Vessel	B-D	B3.90	55%	Configuration of Nozzle Boss
AC-1003	Nozzle to Vessel	B-D	B3.90	55%	Configuration of Nozzle Boss
N2A-IRS	Nozzle Inside Radius	B-D	B3.100	90%	Configuration of Inside Radius
N2B-IRS	Nozzle Inside Radius	B-D	B3.100	90%	Configuration of Inside Radius

Licensee's Basis for Requesting Relief (as stated):

"Relief/Exemption is requested pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii)..., in that the required examination coverage for the identified items is impractical and would require reactor vessel component redesign or replacement to enable the required inspection coverage.

"The Reactor Pressure Vessel (RPV was designed and constructed to ASME Section III, 1965 Edition. This code did not contain requirements to ensure that items be accessible for future examinations. The eight (8) items identified within Attachment Number 1 were installed utilizing this construction code which did not provide for accessibility for future ISI NDE.

"The Class 1 Reactor Pressure Vessel is part of the ASME Section XI VT-2 Leakage Examination boundary. Class 1 Leakage Examinations are performed each refueling outage as required by the Code to insure pressure boundary integrity. In addition to the ASME Section XI leakage examinations, a very sensitive leak detection system, governed by Technical Specification 3.4.15, can detect minimal RCS leakage. The combination of RCS leak detection capability and the Class 1 leakage examination that is performed each refueling outage provide additional assurances in maintaining plant safety.

"All possible means were investigated to increase examination coverage. Radiographic examination is impractical due to the vessel being full of water and high background radiation would mask or prevent meaningful results. Ultrasonic technique variations were employed to maximize examination coverage utilizing latest PDI technology using both 70 degree and 45 degree shear and longitudinal waves from the vessel inside surface. The vessel OD surface was also investigated but this surface is enclosed by a permanent rivetted sheathing and asbestos insulation that has never been removed. To access welds from the outside vessel surface would require scaffolding, building enclosures, removal of the rivetted sheathing, cleaning, disposal, reinstallation of the sheathing and scaffold removal. This major activity would expose work personnel to a high radiation dose of 150 to 200 Rem. There is no current

industry safety concerns associated with these welds and previous examination history show no active degraded conditions exist. No additional coverage beyond the identified coverage in Attachment Number 1 can be achieved that is practical and safe for the work force.

"ASME Section XI Class 1 periodic leakage examinations are performed, and sensitive RCS leak detection capability also exists. This leak detection system and periodic system leakage examinations provide additional assurances in maintaining plant safety. The identified volumetric examination coverage for these items should be acceptable in fulfilling coverage requirements."

Licensee's Proposed Alternative Examination (as stated):

"R.E. Ginna Nuclear Power Plant proposes that the volumetric examination coverage identified within Attachment Number 1 be acceptable in fulfilling required volumetric examination coverage."

Evaluation: The Code requires 100% volumetric examination of the subject RPV nozzle-to-vessel welds and inside radius sections. However, complete examination is restricted due to the nozzle geometries, including nozzle boss and inner radius configurations. These limitations make the 100% volumetric examination impractical. To gain access for examination, the RPV and/or nozzles would require design modifications. Imposition of this requirement would create an undue burden on the licensee.

The licensee has examined a significant portion of these welds, obtaining 55% and 90% coverage for each of the subject nozzle-to-vessel welds and nozzle inside radius sections, respectively. Based on the coverages obtained, it is concluded that any existing patterns of degradation would have been detected by the examinations, and reasonable assurance of the structural integrity has been provided.

Based on the impracticality of meeting the Code coverage requirements for the subject nozzle-to-vessel welds and inside radius sections, and the reasonable assurance provided by the examinations completed, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.4 Request for Relief No. 43, Examination Category C-B, Item C2.32, Residual Heat Removal Heat Exchanger (RHRHX) Nozzle-to-Shell Weld (Third Ten-Year Interval)

Code Requirement: Examination Category C-B, Item C2.32 requires a volumetric examination as defined by Figure IWC-2500-4(c). This examination is specified for nozzle-to-shell (or head) welds with reinforcing plates in vessels greater than ½-inch nominal thickness when the inside of the vessel is accessible for nozzle examinations. Examinations are limited to nozzles at terminal ends of piping runs selected for examination under Examination Category C-F.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% volumetric examination of the Residual Heat Removal Heat Exchanger outlet nozzle welds listed below.

Component ID	Component Description	Estimated %	Description of Limitation
ONSRHE-1B	Outlet Nozzle to Shell	79.0%	Internal welded separation plate
ONSRHE-1A	Outlet Nozzle to Shell	79.0%	Internal welded separation plate

Licensee's Basis for Requesting Relief (as stated):

"Relief is requested pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii), the required examination coverage for the identified items are impractical and would require redesign to allow examination or to be replaced to enable inspection.

"The two identical Residual Heat Removal (RHR) Heat Exchangers was (sic) designed and constructed to ASME Section VIII 1965 Edition. This code did not contain requirements to ensure that items be accessible for future examinations. The two (2) ASME Class 2 items identified above were installed utilizing this construction code which did not provide for accessibility for future ISI NDE. The ISI ASME Section XI volumetric requirement is identified within Table IWC-2500-1, Category C-B, Item Number C2.32.

"The Residual Heat Removal (RHR) Heat Exchangers is part of the ASME Section XI VT-2 Leakage Examination boundary. Leakage Examinations are performed each period as required by Category C-B, Item Number C2.33 of the Code to insure pressure boundary integrity. In addition to the ASME Section XI leakage examinations, Operator walkdowns as specified by Plant Operating Procedures are also performed. The combination of operator walkdowns and period leakage examinations that are performed provide additional assurances in maintaining plant safety."

Licensee's Proposed Alternative Examination (as stated):

"R.E. Ginna Nuclear Power Plant proposes that the volumetric examination coverage identified above be acceptable in fulfilling required volumetric examination coverage."

Evaluation: The Code requires 100% volumetric examination of the subject Residual Heat Removal Heat Exchanger outlet nozzle welds. However, complete examination is restricted due to internal welded separation plates which limit examination coverage. These limitations make the 100% volumetric examination impractical. To gain access for examination, the residual heat removal heat exchanger and/or nozzles would require design modifications. Imposition of this requirement would create an undue burden on the licensee.

The licensee has examined a significant portion of the subject nozzle welds, obtaining 79% coverage for each of the nozzle-to-vessel welds. Based on the coverages that

have been obtained, it is concluded that any existing patterns of degradation would have been detected by the examinations, and reasonable assurance of the structural integrity has been provided.

Based on the impracticality of meeting the Code coverage requirements for the subject nozzle-to-vessel welds, and the reasonable assurance provided by the examinations that were completed, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.5 Request for Relief No. 1 (Revision 1), Examination Category B-A, Items B1.30 and B1.40, Deferral of RPV Shell-to-Flange and Head-to-Flange Weld Examinations (Fourth Ten-Year Interval)

Code Requirement: Examination Category B-A, Items B1.30 requires 100% volumetric examination of the shell-to-flange welds, as defined by Figure IWB-2500-4. Note 3 states that the shell-to-flange weld examination may be performed during the first and third periods, in which case 50% of the shell-to-flange shall be examined by the end of the first period, and the remainder by the end of the third period. During the first period, the examination need only be performed from the flange face, provided this same portion is examined from the shell during the third period.

Item B1.40 requires 100% volumetric and surface examination of the head-to-flange weld, as defined by Figure IWB-2500-5. Note 4 states that deferral is not permissible for the head-to-flange weld. However, during the first and second periods, the examination need only be performed from the flange face, provided these same portions are examined from the head during the third period.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(a)(3), the licensee proposed to defer the subject examination to the end of the interval. The licensee stated:

"R.E. Ginna Nuclear Power Plant proposes that the examinations identified above be performed at or near the end of the interval."

Licensee's Basis for Proposed Alternative (as stated):

"The Reactor Pressure Vessel is a major source of radiation exposure accumulated during a normal refueling outage Inservice inspection. By performing the above Category and Item examinations at the end of the interval there will be no sacrifice in the quantity nor quality of examinations but, there will be a substantial reduction in radiation exposure.

"During the first three inspections intervals, 100 percent of the accessible length of the RPV welds including the Shell-to-Flange weld was examined at or near the end of the interval. The entire examination could be performed from the vessel inside wall from both sides of the weld. This is a more practical approach when performed at the end of the interval using an inside the vessel inspection device.

“The Shell-to-Flange weld is examined with the use of an automated system. To perform this examination requires unloading the core, which results in radiation exposure to workers involved in these fuel transfer activities, an increase of contaminated waste and a risk of a fuel handling accident. Greater examination efficiency with an associated decrease in worker radiation exposure would be obtained when performing 100% of the examination at or near the end of the interval instead of doing 50% in the first period and the remainder in the third period. By eliminating duplication of activities such as staging and testing of equipment, moving of shielding, and other support activities including HP coverage, a reduction of radiation exposure can be realized.

“The Head-to-Flange weld is performed from the head side of the weld and is examined manually when the head is on a stand. Greater examination efficiency with an associated decrease in worker radiation exposure would be obtained when performing this examination at or near the end of the interval instead of each period.

“Performing these examinations once at or near the end of the interval would reduce worker radiation exposure by approximately 1.3 to 1.5 Rem. There are no current industry safety concerns associated with these welds and previous examinations show no active degraded conditions exist. Also, Class 1 Leakage examinations are performed each outage to ensure system/component integrity.

“During the fourth interval, 100 percent of the accessible length of all RPV welds including the Shell-to-Flange and Head-to-Flange welds will be performed at or near the end of the interval when all the required examinations can be performed at the same time.

“In addition the licensee has stated that no inservice repairs have been performed on the subject head to flange and shell to flange weld.”

Evaluation: Examination Category B-A, Item B1.30 requires 100% volumetric examination of the shell-to-flange welds, as defined by Figure IWB-2500-4. Note 3 states that the shell-to-flange weld examination may be performed during the first and third periods, in which case 50% of this weld shall be examined by the end of the first period, and the remainder by the end of the third period. During the first period, the examination need only be performed from the flange face, provided this same portion is examined from the shell during the third period. Item B1.40 requires 100% volumetric and surface examination of the head-to-flange weld, as defined by Figure IWB-2500-5. Note 4 states that deferral is not permissible for the head-to-flange weld. However, during the first and second periods, the examination need only be performed from the flange face, provided these same portions are examined from the head during the third period. The licensee has proposed to defer the subject examinations to the end of the interval.

The proposed alternative contained in this request for relief parallels that described in Code Case N-623, *Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds*

of a Reactor Vessel, which allows for deferral of volumetric examinations of the reactor pressure vessel shell-to-flange and head-to-flange welds to the end of the interval.

Code Case N-623 has been approved for use at other nuclear facilities when the conditions contained in the code case have been met. The following three conditions must be met in order to defer the shell-to-flange and head-to-flange welds to the end of the inspection interval:

(1) No welded repair/replacement activities have ever been performed on the shell-to-flange or head-to-flange weld.
(2) Neither the shell-to-flange weld nor head-to-flange weld contains identified flaws or relevant conditions that currently require successive inspections in accordance with IWB-2420(b).
(3) The vessel is not in the first inspection interval.

The licensee stated that no inservice repair activities have been performed on the shell-to-flange or head-to-flange welds. However, repair (by removal of in-process welding flaws) occurred on the subject welds during the normal course of initial construction of the reactor pressure vessel. The INEEL interprets condition (1) above to be associated with inservice repair/replacement activities performed during the current license period only and not during initial construction. Therefore, condition (1) is satisfied for the Ginna RPV. Further, the licensee has stated that no relevant conditions exist that require successive inspections for these welds, and Ginna is not in the first inspection interval. The INEEL concludes that the conditions specified by Code Case N-623 have been met.

Industry experience to date indicates that examinations performed on reactor pressure vessel shell-to-flange and head-to-flange welds have not identified any detrimental flaws or relevant conditions that might impact the structural integrity of these welds. Also, changing the schedule for examining these welds to the end of the licensee's 10-year inservice inspection interval provides a suitable frequency for verifying the integrity of the subject welds[2]. Considering that volumetric examinations of these welds at Ginna, and similar welds at other plants, have resulted in no detrimental flaws or relevant conditions, and that volumetric examinations will continue to be performed on an interval basis, it is determined that the licensee's proposed alternative, to defer the subject examinations, provides an acceptable level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i).

2.6 Request for Relief No. 2, Examination Category B-M-2, Item B12.50, Internal Surfaces of Valve Bodies (Fourth Ten-Year Interval)

Code Requirement: Examination Category B-M-2, Item B12.50 requires a VT-3 visual examination of the internal surfaces of valve bodies. The examinations are limited to at least one valve within each group of valves that are of the same size, constructional

2. Information obtained from NRC Safety Evaluation Report for Millstone Nuclear Power Station Unit 2, dated March 15, 2000.

design (such as globe, gate, or check valve) and manufacturing method, and that perform similar functions in the system. Note 2 states that examination is required only when a pump or valve is disassembled for maintenance, repair, or volumetric examination.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed to perform VT-3 examinations of valve bodies when they are disassembled. The licensee stated:

"As stated above RG&E does not believe that the visual examination required each ten-year interval is warranted. However, 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 Proposed Alternative (as stated):

"This relief request is based on the following points:

1. "To complete the subject examination, unnecessary expenditures of man-hours and manrem are required with essentially no compensating increase in plant safety, and

2. "The structural integrity afforded by valve casing material utilized 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 man-hours and 5-manrem exposure would be required to disassemble, inspect, and reassemble these valves. Performing this visual examination under such adverse conditions, high doses rate (30-40 R/hr), and poor as-cast surface conditions, realistically provides little additional information as to the valve's casing integrity. The valves' 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 and corrosion in chloride containing environments.

"RG&E feels that adequate safety margins are inherent in the basic valve design and that the public's 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. Therefore, a request for relief from this requirement is sought. A VT-3 examination shall be performed once on one valve within the valve group during the Interval if disassembled for maintenance.

" Also, Class 1 Leakage examinations are performed each outage to ensure system/component integrity."

Evaluation: Examination Category B-M-2, Item B12.50 requires a VT-3 visual examination of the internal surfaces of valve bodies. The examinations are limited to at least one valve within each group of valves that are of the same size, constructional design (such as globe, gate, or check valve) and manufacturing method, and that perform similar functions in the system. Note 2 of Examination Category B-M-2 states that examination is required only when a valve is disassembled for maintenance, repair, or volumetric examination.

The Code recognizes that disassembly of the valves for the sole purpose of inspection is a major effort and, in addition to the possibility of damage that may occur to the valves, could result in personnel receiving excessive radiation exposure, as is described by the licensee. The Code takes this into consideration as evidenced in Note 2 of Examination Category B-M-2, by requiring examination only when a pump or valve is disassembled for *maintenance, repair, or volumetric examination*.

The licensee has proposed to perform the visual examination when the subject valves are disassembled for maintenance purposes. Based upon the licensee's proposed alternative, it appears the Code requirements are being met. Therefore, relief is not required.

2.7 Request for Relief No. 3, Examination Category B-J, Reactor Coolant Cast Pump Terminal End-to-Cast Elbow Circumferential Welds (Fourth Ten-Year Interval)

Code Requirement: Examination Category B-J, Item B9.11 requires 100% surface and volumetric examinations on pressure retaining welds in piping 4" NPS or larger each inspection interval as defined by Figure IWB-2500-8.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required volumetric examinations of the following Reactor Coolant Pump terminal end-to-cast elbow circumferential welds.

Weld ID	Type
PL-FW-XIII	Elbow-to-Pump
PL-FW-XV	Elbow-to-Pump

Licensee's Basis for Requesting Relief (as stated):

"Supplement 4 (b)(4) of Appendix III identifies ultrasonic scanning requirements on Austenitic and Dissimilar Metal Welds. This paragraph further identifies that Cast items such as fittings, valve bodies, and pump casings may preclude meaningful examinations because of geometry and attenuation variables.

“At R. E. Ginna Nuclear Power Plant, the Reactor Coolant Pump, Westinghouse Model 93, is cast stainless (A-351 Gr CF8M). The associated fittings (elbows) are also cast stainless (ASTM A-351 Gr CF8M) and contains longitudinal seam welds. The Reactor Coolant Inlet/Outlet Wrought Safe End/Pipe to Elbow circumferential welds consist of fittings (elbows) that are cast stainless (ASTM A-351 Gr CF8M). When employing optimized ultrasonic techniques on these cast/wrought welds, the techniques may detect large flaws (25% or greater through wall). Therefore, the sensitivity is less than that required by the Code.

“Cast Stainless base metal and associated welds contain large grain structures, attenuation variables impact the performance of ultrasonic examinations. Experience has shown that these materials are not always amenable to ultrasonic examination and does not produce reliable and meaningful results. Currently, the industry's Performance Demonstration Initiative (PDI) is not addressing Cast Stainless Ultrasonic Examinations.

“Due to the highly attenuative characteristics of the austenitic grain structure, ultrasonic examination coverage to the extent that is specified within Code Case N-460 may not always be achievable.

“Radiography, if applied, is not expected to provide any meaningful increase in benefit beyond the alternative presented due to the high levels of background radiation emitting from these areas, significantly decreasing the signal (image to noise ratio) of the radiograph.”

Licensee’s Proposed Alternative Examination (as stated):

“None. Applicable Code-required volumetric examination (UT) will be completed to the maximum extent practical (a best effort examination of the cast stainless welds based on state-of-the-art techniques and associated achievable examination coverage). We will continue to evaluate new emerging inspection technology as they become available. The Code required surface examinations and system leakage tests will be performed.”

Evaluation: The Code requires 100% volumetric examination of the subject reactor coolant pump-to-cast pipe elbow welds. However, the acoustical properties of the cast stainless steel components (A-351 Grade CF8M) make the Code coverage requirements impractical for these welds. Due to the highly attenuative characteristics of the austenitic grain structure, ultrasonic examination coverage to the extent that is specified by the Code (from both sides of the weld) is not achievable.

For the subject Item B9.11 welds, the licensee can perform 45-50% of the Code-required volumetric examinations and 100% of the surface examinations. In addition, other B-J Category welds will be examined to the Code requirements. Based upon the limited portion of the welds examined volumetrically, the 100% surface examination coverage, and the completion of other welds in this Code Category, it is concluded that significant patterns of degradation would be detected, and reasonable assurance of the structural integrity of these welds will be provided.

Based on the impracticality of meeting the Code requirements for the subject examination areas, and the reasonable assurance provided by the examinations that will be completed, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i) for the subject welds.

2.8 Request for Relief No. 4, Examination Categories B-B and B-D, Items B2.60, B2.80, and B3.150, Regenerative Heat Exchanger (RHE) Welds and Nozzle Inside Radius Sections (Fourth Ten-Year Interval)

Code Requirement: Examination Category B-B, Items 2.60 and 2.80 require a 100% volumetric examination as defined by figure IWB-2500-6 for primary side heat exchanger tubesheet-to-shell and tubesheet-to-head welds. Examination Category B-D, Item B3.150 requires 100% volumetric examination as defined by Figure IWB-2500-7 for full penetration welds of nozzles on the primary side of Class 1 heat exchangers.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed to perform volumetric examination on only one of three identical Class 1 Regenerative Heat Exchangers (RHE) sections. The licensee stated:

"Rochester Gas & Electric (RG&E) proposes to utilize the 'multiple stream' concept when performing a volumetric examination of accessible portions of Nozzle-to-Vessel Welds and Nozzle Inside Radius Welds equivalent to one of the three identical sections on the Class 1 Regenerative Heat Exchanger. Examinations shall be performed on the lower section. The associated examinations shall be performed once during the interval. In addition, RG&E proposes to perform a VT-2 visual examination on the entire Regenerative Heat Exchanger during system leakage tests."

Licensee's Basis for Proposed Alternative (as stated):

"The Regenerative Heat Exchanger (RHE) consists of three (3) shell and tube heat exchangers connected in series. The RHE is designed to recover heat from the reactor coolant system letdown stream during normal operation. The letdown stream flows through the shell side of the heat exchanger. The shell side of the RHE is Class 1 while the tube side is Class 2.

"The Regenerative Heat Exchanger provides the major single source of radiation exposure accumulated during a normal refueling outage Inservice inspection. By performing the above Category and Item number examinations on one of the three heat exchangers seeing the most extreme conditions will provide a representative sample, while limiting personnel radiation exposure.

"For the above Categories & Item Numbers, the bottom heat exchanger should be the one heat exchanger selected since it operates at the highest temperature of all the units and is therefore the most highly stressed. Typical operating temperatures for letdown flow is around 544 Deg. F. and between 300-350 Deg. F. out of top shell. By limiting the examinations to one heat exchanger will significantly reduce radiation exposure to personnel. The exposure savings to

ISI related personnel per inspection interval would be 19.6 Man-Rem Whole Body and 68.0 Man-Rem Extremities through the reduction of 10 examinations."

Evaluation: The Code requires 100% volumetric examination of the subject RHE welds. The licensee has requested relief from examining two of the three sections of the RHE, and has proposed to examine the equivalent of one section as a representative sample for all three portions of the RHE. The licensee considers the RHE to be multiple vessels, each performing a similar function, and has proposed to apply the multiple stream concept and examine the welds of only one vessel. The licensee also stated that the RHE is a significant source of radiation exposure and estimates that personnel exposure can be reduced by 19.6 man-Rem whole body and 68.0 man-Rem extremities through the reduction of 10 examinations.

The three sections of the RHE are of similar design and function. However, it appears from the licensee's basis that the three RHE sections do not operate at equal conditions (i.e. different temperatures and stresses). Therefore, the multiple stream concept is not necessarily applicable. Nevertheless, the RHE is a significant radiation source; personnel exposure can be substantially reduced by limiting examinations to just one section of the RHE. The section selected for examination under the licensee's proposed alternative experiences the highest temperatures and stresses (the lowermost of the three vessel sections). Since the vessel section subjected to the most severe operating conditions will be examined, patterns of degradation that may be present are more likely to be detected in this portion of the RHE. Therefore, reasonable assurance of the structural integrity of all RHE welds will be provided.

Considering the potential radiation exposure that would be associated with examining all three sections of the RHE, and the reasonable assurance provided by examining the vessel section that experiences the most severe conditions, it is concluded that compliance with the Code requirements would result in a hardship without a compensating increase in safety. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55(a)(3)(ii).

2.9 Request for Relief No. 5, Alloy 690 Weld Material, Fabrication of New Steam Generators, Use of Code Case 2142, *F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal,* and Code Case 2143, *F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode* (Fourth Ten-Year Interval)

Code Requirement: IWA-4223.3, requires:

(a) Use of materials of a specification, grade, type, class, or alloy, and heat-treated condition, other than that originally specified, shall be evaluated for suitability for the specified design and operating conditions in accordance with IWA-4311.
(b) Material examination and testing requirements shall be reconciled to the Construction Code requirements of the component.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed to incorporate the use of Code Cases 2142, and 2143. The licensee stated:

"Incorporate ASME, Boiler & Pressure Vessel Code Case 2142, 'F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal', and Case 2143, 'F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode'."

Licensee's Basis for Proposed Alternative (as stated):

"The use of Alloy 690 type weld filler material is required for the replacement steam generators. These materials have been approved by ASME through Code Cases 2142 and 2143 and are designated as UNS N06052 and UNS W86152, respectively, and classified them as F-No. 43 for weld procedure and performance qualification purposes in accordance with ASME Section XI.

"UNS W86152 is the shielded metal arc welding electrode for Alloy 690 and UNS N06052 is the bare filler metal. Both materials have been shown in numerous EPRI studies to have improved corrosion resistance for Alloy 690 weldments as compared to the currently used Ni-Cr-Fe(N06082 and W86182) materials. The new weld materials are the preferred choice for welding applications involving Alloy 690 in a corrosion environment and they provide an acceptable level of quality and safety because of their superior corrosion resistant properties."

Evaluation: IWA-4223.3, requires that:

(a) Use of materials of a specification, grade, type, class, or alloy, and heat-treated condition, other than that originally specified, shall be evaluated for suitability for the specified design and operating conditions in accordance with IWA-4311.
(b) Material examination and testing requirements shall be reconciled to the Construction Code requirements of the component.

The licensee has requested the use of Code Cases 2142, *F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal*, and 2143, *F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode*, for use on fabrication and installation replacement steam generators.

The following information is taken from the licensee's third interval Safety Evaluation Report (SER), performed by the NRC and dated September 11, 1995.

Due to Intergranular Stress Corrosion Cracking (IGSCC) problems associated with alloy 600, the Nuclear industry has pursued the use of an alternative alloy. Currently, alloy 690 is the industry material utilized in the fabrication and installation of replacement Steam Generators. The use of this alloy has been based on numerous laboratory studies which show that alloy 690 virtually has no susceptibility to IGSCC in environments that simulate pressurized-water reactor (PWR) and boiling-water reactor (BWR) conditions. The NRC staff performed a review of these test results and determined that no technical reason exists to disallow the use of alloy 690 base material in nuclear plant construction[3].

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3. Information obtained in Safety Evaluation Report dated 09/11/95

Alloy 600 type weld metals (such as Inco 82 and 182) were widely used during the construction of nuclear power plants. Operating experience showed that Inco 182 was also susceptible to IGSCC, although primarily in BWR environments. Weld metals matching alloy 690 have also been tested in simulated PWR and BWR environments. Commercial development of these weld metals lagged that of the alloy 690 base metal.

Corrosion studies examining the susceptibility of weld metals to IGSCC in SG environments are scant compared to the voluminous base metal studies. This is because the base metal performance is a strong indicator of the expected performance of a matching weld metal. Results of the principal study which included weld metals are found in the Electric Power Research Institute (EPRI) report NP-5882M, titled, "Stress Corrosion Cracking Resistance of Alloys 600 and 690 and Compatible Weld Metals in BWR's." Two experimental alloy 690 weld metals were tested. They were tested under the same conditions as the base metals; thus, allowing direct comparison of results. Results showed that both of the alloy 690 weld metals are immune to IGSCC in pure water environments. However, since these were laboratory simulations of a BWR environment, the results are only an indicator, and not a guarantee, of the weld metals performance in a PWR environment.

In this report the designations R-127 and R-135 were used for the experimental weld metals. These were the Inco designations for the development weld metals that became Inco 52 and 152, respectively. Inco 52 is the commercial filler metal (tig wire) described in ASME Code Case 2142. Inco 152 is the coated electrode described in Code Case 2143.

Another paper, "Inconel 690; A New High Nickel Alloy for Corrosive Environments at Elevated Temperature," by A. J. Sedricks, et. al. of the Inco Research and Development Center, included tests of a matching filler metal in a wide variety of environments. The two most interesting tests were conducted in simulated SG environments: deaerated ammoniated and borated water at 316 degrees C. Test results showed the welds and weld metal were highly resistant to general corrosion.

Stress-corrosion cracking susceptibility was tested by exposing welds to a variety of chloride environments. The controls used in these tests were alloy 800 (not 600) and type 304 stainless steel. Both of these alloys are known to crack in elevated temperature chloride environments. In all cases, alloy 690 was tested for periods significantly longer than the time to crack alloy 800 (the more resistant of the two control alloys). In no case did the alloy 690 welds crack despite test durations 8 times longer than that of the control alloys.

Additional testing for IGSCC susceptibility in pure water environments was conducted. Another group of alloy 690 welds plus control alloys were exposed to undeaerated water at elevated temperatures in the presence of a crevice. Cracking was readily initiated within the controls. None of the alloy 690 welds cracked despite testing durations 24 times longer than for alloy 600 and 12 times for alloy 800 and 304 stainless.

Not considered in either study is the effect, if any, of heat to heat variations in the weld metal compositions. Such variations were found to play a substantial role in the IGSCC

susceptibility of alloy 600. The strong performance of alloy 690 suggests there would be minimal effect.

ASME Code Case 2142 lists the American Welding Society (AWS) specification (AWS A5.14) and UNS designation (UNS N06052) for filler metal conforming to INCO 52. It establishes the F-No. of this weld metal as F-No. 43 for both procedure and performance qualification purposes. Code Case 2143 lists appropriate AWS and UNS specifications for a coated electrode matching Inco 152 and establishes F-No. 43 for this material for welding purposes. By this set of specifications and F-No. assignments, these materials are completely described for welding purposes as similar in their welding characteristics to many other Code nickel base weld metals. These materials are similar in composition and mechanical/thermal properties to approved metals; thus, these two weld metals need not meet the requirements for specific procedure and performance qualifications for non-Code materials.

Therefore, considering that this proposed alternative was approved during the third ten year interval, and the technical justification for the approval remains intact, it is recommended that the proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i).

2.10 Request for Relief No. 6, Examination Category B-D, Item B3.160, Regenerative Heat Exchanger (RHE), Inner Radius Examinations (Fourth Ten-Year Interval)

Code Requirement: Examination Category C-D, Item B3.160 requires 100% volumetric examination, as defined by Figure IWB-2500-7, for heat exchanger nozzle inner radius (IR) sections.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code requirements for the Regenerative Heat Exchanger (RHE) welds listed below:

RHE-N1
RHE-N5

Licensee's Basis for Requesting Relief (as stated):

"The Regenerative Heat Exchanger (RHE) consists of three (3) shell and tube heat exchangers connected in series. The RHE is designed to recover heat from the reactor coolant system letdown stream during normal operation. The letdown stream, flows through the shell side of the heat exchanger. The shell side of the RHE is Class 1 while the tube side is Class 2.

"The RHE provides the major single source of radiation exposure accumulated during a normal refueling outage Inservice inspection. Inner Radius examinations were scheduled to be performed on the Class 1 side of the bottom heat exchanger that experience the most extreme conditions as specified within Relief Request #4.

“The RHE inner radius examinations were to be performed on small heavy wall nozzles that are connected to 2" piping. An in-depth investigation was initiated by RG&E to determine the feasibility of performing an acceptable Code examination. The initial investigation reviewed the nozzle type, weld placement and actual OD weld profiles as well as ultrasonic measurements to verify ID configuration.

“To assist in the evaluation of performing an acceptable Code ultrasonic inner radius examination, Computer Modeling and mockups of the nozzle to vessel configuration were initiated. Computer modeling was performed by Southwest Research Institute, AEA Technology, and EPRI.

“Computer modeling performed by the various organizations compared favorably. The computer modeling initiative indicated several different transducers would be required to be used and the inner radius examination results would be questionable at best due to the size and configuration of the nozzles. The modeling also indicated beam spread and mode conversion at the notches and neighboring surfaces would seriously reduce the signal to noise ratios, causing confusing spurious signals.

“Based upon the computer modeling results, EPRI NDE Center personnel were utilized to perform actual hands-on inner radius examination evaluation on the mockups. An area was selected on two mockups of the nozzle, suitable transducers and wedges were selected to perform the examination. The inspection was to be performed from the boss region of the nozzle because inspection from the shell surface proved to be greatly affected by attenuation and scattering from the nozzle-to-shell weld material. A variety of inspection frequencies were attempted of which none provided what would be considered successful for the detection of the notches on these nozzles. It should be noted that the transducer position for detecting the selected notch was nearly optimum. Since the attempts made were unsuccessful it was decided to increase the depth of the notch from 10% to 30%. This increase of the notch is greater than Code allowable. Attempts were made on the greater notch depth but detection was not achievable.

Licensee’s Proposed Alternative Examination (as stated):

“None, RG&E will continue to evaluate new emerging inspection technology as they become available. The Code required leakage tests with associated VT-2 examinations shall be performed.”

Evaluation: The Code requires 100% volumetric examination of the subject nozzle inner radius sections. However, complete examination of these areas is limited by nozzle/inner radius configuration. Due to the nozzle/inner radius geometry, the licensee initiated computer modeling to assist in the evaluation of performing an acceptable Code inner radius examination. In addition, mockups of the configuration including artificial flaws (notches) were fabricated. EPRI NDE personnel were employed to perform actual hands-on inner radius examinations of the mockups. Multiple techniques were utilized,

however, none of the methods used provided satisfactory detection results of the notches. Based upon the computer modeling and mockup test results, it was determined that the Code examination requirements are impractical. In order to meet the Code requirements, design modifications of the subject components would be necessary, resulting in an undue hardship on the licensee.

Therefore, based upon the impracticality of examining the inner radius sections, the licensee's commitment to continue to evaluate new emerging inspection technologies as they become available, and the licensee's performance of the Code required leakage tests with associated VT-2 examinations, reasonable assurance of the continued structural integrity will be provided. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.11 Request for Relief No. 7, Examination Category C-C, Item C3.20, Integrally Welded Attachments to Piping (Fourth Ten-Year Interval)

Code Requirement: Examination Category C-C, Item C3.20 requires a 100% surface examination of the integrally welded attachments on Class 2 piping as defined by Figure IWC-2500-5.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required surface examinations of the following Class 2 integral attachment welds.

Support Number	Coverage Obtained	System
MSU-33	50%	Main Steam (MS)
MSU-34	50%	Main Steam (MS)
Penetration 140	82%	Residual Heat Removal (RHR)

Licensee's Basis for Requesting Relief (as stated):

"Relief is requested pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii), the required examination coverage for the identified items are impractical and would require redesign to allow examination or to be replaced to enable inspection.

"R. E. Ginna Nuclear Power Plant was designed and constructed to the B31.1, 1955 edition construction code. This code did not contain requirements to ensure that items be accessible for future examinations. The above noted integral attachments associated with the component supports or penetration anchor supports were installed utilizing this construction code which did not provide for accessibility for future ISI NDE. Due to the limited design accessibility, ISI examinations cannot be performed on the inaccessible welds.

"The two Main Steam supports (MSU-33 and MSU-34) are similar in design in that [they] have a complex gusset assembly that is welded to the process piping (integral attachment) and welded to the base plate, which is secured to the

concrete floor. Due to the small size of gussets, access is limited for both the surface examination and surface preparation of the integral attachment welds located under the process piping. The achievable access percentage has been identified above within this relief request.

"The identified component supports (including the integral attachments) are periodically visually examined (VT-3). ASME Section XI periodic leakage examinations are performed as well as Operator walkdowns as specified by Plant Operating Procedures. These Operator walkdowns, periodic system leakage examinations and component support visual examinations provide additional assurances in maintaining plant safety."

Licensee's Proposed Alternative Examination (as stated):

"R. E. Ginna Nuclear Power Plant proposes that the surface examination coverage identified above be acceptable in meeting Code Requirements and is not conducive in obtaining the requirements specified within Code Case N-460."

Evaluation: The Code requires 100% surface examination of the subject integrally welded attachments. Ginna Nuclear Power plant was designed and constructed to the 1955 Edition of ANSI B31.1 and accessibility for examination was not a design consideration. Review of the documentation submitted by the licensee revealed that complete examination coverage is impractical due to restricted access caused by interference from complex gusset assemblies. To meet the Code requirements, the integral attachments and/or interfering structures would require design modification and/or removal to allow access to the subject attachments. Therefore, surface examination of the subject integral attachment welds is impractical to perform to the extent required by the Code. Imposition of this requirement would create a considerable burden on the licensee.

The licensee has completed a significant portion ranging from 50% to 82% of the Code-required surface examinations for the subject components. Based upon the surface coverage obtained for each component, it is reasonable to conclude that existing patterns of degradation, if present, would have been detected, thus providing reasonable assurance of the structural integrity of the subject integral attachment welds. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.12 Request for Relief No. 8, Examination Category B-F, Item B5.10, Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles (Fourth Ten-Year Interval)

Code Requirement: Examination Category B-F, Item B5.10, requires 100% volumetric and surface examination, as defined by Figure IWB-2500-8, of the dissimilar metal Reactor Pressure Vessel nozzle-to-safe end butt welds.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required surface examination coverage for the following Reactor Pressure Vessel nozzle-to-safe end butt welds.

Weld ID #	Coverage Obtained
PL-FW-II	62%
PL-FW-V	75%
PL-FW-IV	68.5%
PL-FW-VII	72%
AC-1003-1	0%*
AC-1002-1	0%*
* welds embedded in concrete	

Licensee's Basis for Requesting Relief (as stated):

"R. E. Ginna Nuclear Power Plant was designed and constructed to the B31.1, 1955 edition Construction code. This code did not contain requirements to ensure that items be accessible for future examinations. The above noted piping welds were installed utilizing this construction code, which did not provide for accessibility for future ISI NDE. Due to the limited design accessibility, ISI surface examination coverage is below Code percentage requirements as identified within this Relief Request.

"The first four (4) welds of this Relief Request are located in a 'Sandbox' configuration. Within the 'Sandbox', the welds are against the floor and one wall. The angled wall is joined to the floor and is against the weld. The surface examination of these welds is limited due to Original Construction Code interferences of the floor and wall of the 'Sandbox'. The 'Sandboxes' would have to be redesigned to enable the welds to be inspected to obtain Code required coverage for the surface examinations. The last two (2) welds of this Relief Request are embedded in concrete. This concrete structure is the wall that surrounds the Reactor Pressure Vessel.

"ASME Section XI Class 1 system leakage examinations are performed. These leakage examinations demonstrate pressure boundary integrity and provide additional assurances in maintaining plant safety."

Licensee's Proposed Alternative Examination (as stated):

"R. E. Ginna Nuclear Power Plant proposes that the surface examination coverage identified for the first four (4) welds above be acceptable in fulfilling the Code required examination coverage. The actual physical configuration of the 'Sandboxes' is not conducive in obtaining the requirements specified within Code Case N-460 for acceptable coverage. Volumetric examination of these welds is performed from the inside of the Vessel, and will be performed during the 2009 outage.

"For the last two (2) welds, the Code surface examination requirements are impractical and cannot be examined due to them being embedded in concrete. Volumetric examination of these welds is performed from the inside of the Vessel, and will be performed during the 2009 Outage."

Evaluation: The Code requires 100% volumetric and surface examination of the subject Reactor Pressure Vessel nozzle-to-safe end butt welds. Ginna Nuclear Power plant was designed and constructed to the 1955 Edition of ANSI B31.1 and accessibility for examination was not a design consideration. Review of the documentation submitted by the licensee demonstrates that complete surface examination coverage is impractical due to restricted access caused by interference from the containment floor and wall, or encasement in concrete. To meet the Code requirements, the subject nozzles and/or interfering structures would require design modification and/or removal to allow access to the subject welds. Therefore, the surface examination of the subject welds is impractical to perform to the extent required by the Code. Imposition of this requirement would create a considerable burden on the licensee.

The licensee can complete up to 75% of the Code required surface examinations on the subject welds, with exception of those encased in concrete. In addition, the licensee performs 100% of the required volumetric examination of the subject welds (these examinations are performed from the ID of the pipe). Based upon the limited surface coverages achievable, and the 100% volumetric examinations performed, it is reasonable to conclude that existing patterns of degradation, if present, will be detected, providing reasonable assurance of the structural integrity of the welds. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.13 Request for Relief No. 14 (Part A), Examination Category B-A, Items B1.11, and B1.30, Pressure-Retaining Welds in Reactor Vessels (Fourth Ten-Year Interval)

Code Requirement: Examination Category B-A, Item B1.11 and B1.30 require 100% volumetric examination of the Reactor Pressure Vessel Circumferential Shell Weld and Shell-to-Flange Weld, as defined by Figures IWB-2500-1 and IWB-2500-4 respectively.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% volumetric examination of the welds listed below.

Comp. ID	Component Description	ASME Category	Item Number	Coverage	Limitation
RPV-A	Vessel to Flange Weld	B-A	B1.30	54%	Obstructed by keyways and irradiation slots
RPV-D	Lower shell to ring forging	B-A	B1.11	81%	Obstructed by guide lugs and incore instrumentation

Licensee's Basis for Requesting Relief (as stated):

"Relief is requested pursuant to the provisions of 10 CFR 50.55a (g)(5)(iii), the required examination coverage for the identified items are impractical and would require redesign to allow examination or to be replaced to enable inspection.

"The Reactor Pressure Vessel (RPV) was designed and constructed to ASME Section III, 1965 Edition. This code did not contain requirements to ensure that items be accessible for future examinations. The eight (8)[4] items identified within Attachment Number 1 was (sic) installed utilizing this construction code which did not provide for accessibility for future ISI NDE.

"The Class 1 Reactor Pressure Vessel is part of the ASME Section XI VT-2 Leakage Examination boundary. Class 1 Leakage Examinations are performed each refueling outage as required by the Code to insure pressure boundary integrity. In addition to the ASME Section XI leakage examinations, Operator walkdowns as specified by Plant Operating Procedures are also performed. The combination of operator walkdowns and the Class 1 leakage examination that is performed each refueling outage provide additional assurances in maintaining plant safety."

Licensee's Proposed Alternative Examination (as stated):

"R. E. Ginna Nuclear Power Plant proposes that the volumetric examination coverage identified within Attachment[5] Number 1 be acceptable in fulfilling required Code volumetric examination coverage."

Evaluation: The Code requires that the subject reactor pressure vessel welds be 100% volumetrically examined during the inspection interval. Due to access limitations resulting from keyways, irradiation slots, guide lugs and in-core instrumentation, examinations are limited to 54% and 81%, respectively, of the required volume of the subject welds. Based on the information provided in this request for relief, it is concluded that examination of the subject welds to the extent required by the Code is impractical. For complete examination, redesign and modification of the reactor vessel would be necessary. Imposition of this requirement would cause a considerable burden on the licensee.

Based upon the volumetric examination of the accessible portion of the subject welds, and the volumetric examinations on other RPV welds, it is concluded that significant patterns of degradation, if present, will be detected. As a result, reasonable assurance

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4. The total of 8 eight items include Part A and Part B of this request for relief

5. Drawings, sketches and attachments have not been included with this report.

of continued structural integrity will be provided. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted.

2.14 Request for Relief No. 14 (Part B), Examination Category B-D, Full Penetration Welds of Nozzles in Vessels (Fourth Ten-Year Interval)

Code Requirement: Examination Category B-D, Items B3.90 and B3.100, require 100% volumetric examination of nozzle-to-reactor vessel welds and nozzle inside radius sections, as defined by Figure IWB-2500-7.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% volumetric examination of the reactor vessel nozzle welds listed below.

Comp. ID	Component Description	ASME Category	Item Number	Coverage	Limitation
N1A	Nozzle to Vessel	B-D	B3.90	55%	Configuration of Nozzle Boss
N1B	Nozzle to Vessel	B-D	B3.90	55%	Configuration of Nozzle Boss
AC-1002	Nozzle to Vessel	B-D	B3.90	55%	Configuration of Nozzle Boss
AC-1003	Nozzle to Vessel	B-D	B3.90	55%	Configuration of Nozzle Boss
N2A-IRS	Nozzle Inside Radius	B-D	B3.100	90%	Configuration of Inside Radius
N2B-IRS	Nozzle Inside Radius	B-D	B3.100	90%	Configuration of Inside Radius

Licensee's Basis for Requesting Relief (as stated):

"Relief is requested pursuant to the provisions of 10 CFR 50.55a (g)(5)(iii), the required examination coverage for the identified items are impractical and would require redesign to allow examination or to be replaced to enable inspection.

"The Reactor Pressure Vessel (RPV) was designed and constructed to ASME Section III, 1965 Edition. This code did not contain requirements to ensure that items be accessible for future examinations. The eight (8)[6] items identified within Attachment Number 1 was (sic) installed utilizing this construction Code which did not provide for accessibility for future ISI NDE.

"The Class 1 Reactor Pressure Vessel is part of the ASME Section XI VT-2 Leakage Examination boundary. Class 1 Leakage Examinations are performed each refueling outage as required by the Code to insure pressure boundary

6. The total of 8 eight items include Part A and Part B of this request for relief

integrity. In addition to the ASME Section XI leakage examinations, Operator walkdowns as specified by Plant Operating Procedures are also performed. The combination of operator walkdowns and the Class 1 leakage examination that is performed each refueling outage provide additional assurances in maintaining plant safety."

Licensee's Proposed Alternative Examination (as stated):

"R. E. Ginna Nuclear Power Plant proposes that the volumetric examination coverage identified within Attachment Number 1 be acceptable in fulfilling required Code volumetric examination coverage."

Evaluation: The Code requires 100% volumetric examination of the subject RPV nozzle-to-vessel welds and inside radius sections. However, complete examination is restricted due to nozzle geometries including nozzle boss and inner radius configurations. These limitations make the 100% volumetric examination impractical. To gain access for examination, the RPV and/or nozzles would require design modifications. Imposition of this requirement would create an undue burden on the licensee.

The licensee has examined a significant portion of these welds, obtaining 55-90% coverage for each of the nozzle-to-vessel welds and nozzle inside radius sections. Considering the coverages that can be obtained, it is concluded that any existing patterns of degradation will be detected by the examinations that can be completed and reasonable assurance of the structural integrity will be provided.

Based on the impracticality of meeting the Code coverage requirements for the subject nozzle-to-vessel welds and inside radius sections, and the reasonable assurance provided by the examinations that will completed, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.15 Request for Relief No. 15, Examination Category C-B, Item C2.32, Residual Heat Removal Heat Exchanger Nozzle-to-Shell Weld (Fourth Ten-Year Interval)

Code Requirement: Examination Category C-B, Item C2.32 requires a volumetric examination as defined by Figure IWC-2500-4(c). This examination is specified for nozzle-to-shell (or head) welds with reinforcing plates in vessels greater than ½-inch nominal thickness when the inside of the vessel is accessible for nozzle examinations. Examinations are limited to nozzles at terminal ends of piping runs selected for examination under Examination Category C-F.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% volumetric examination of the Residual Heat Removal Heat Exchanger (RHRHX) outlet nozzle welds listed below.

Component ID	Component Description	Coverage %	Description of Limitation
ONSRHE-1B	Outlet Nozzle to Shell	79.0%	Internal welded separation plate

Component ID	Component Description	Coverage %	Description of Limitation
ONSRHE-1A	Outlet Nozzle to Shell	79.0%	Internal welded separation plate

Licensee's Basis for Requesting Relief (as stated):

"Relief is requested pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii), the required examination coverage for the identified items are impractical and would require redesign to allow examination or to be replaced to enable inspection.

"The two identical Residual Heat Removal (RHR) Heat Exchangers was (sic) designed and constructed to ASME Section VIII, 1965 Edition. This code did not contain requirements to ensure that items be accessible for future examinations. The two (2) ASME Class 2 items identified above were installed utilizing this construction code which did not provide for accessibility for future ISI NDE. The ISI ASME Section XI volumetric requirement is identified within Table IWC-2500-1, Category C-B, Item Number C2.32.

"The Residual Heat Removal (RHR) Heat Exchangers is part of the ASME Section XI VT-2 Leakage Examination boundary. Leakage Examinations are performed each period as required by Category C-B, Item Number C2.33 of the Code to insure pressure boundary integrity. In addition to the ASME Section XI leakage examinations, Operator walkdowns as specified by Plant Operating Procedures are also performed. The combination of operator walkdowns and period leakage examinations that are performed provide additional assurances in maintaining plant safety."

Licensee's Proposed Alternative Examination (as stated):

"R.E. Ginna Nuclear Power Plant proposes that the volumetric examination coverage identified above shall be acceptable in fulfilling required volumetric examination coverage."

Evaluation: The Code requires 100% volumetric examination of the subject RHRHX outlet nozzle welds. However, complete examination is restricted due to internal welded separation plates which limit access for coverage. These limitations make the 100% volumetric examination impractical. To gain access for examination, the residual heat removal heat exchanger and/or nozzles would require design modifications. Imposition of this requirement would create an undue burden on the licensee.

The licensee has examined a significant portion of the subject nozzle welds, obtaining 79% coverage for each of the nozzle-to-vessel welds. Considering the coverages that have been obtained, it is concluded that any existing patterns of degradation would have been detected by the examinations and reasonable assurance of the structural integrity has been provided.

Based on the impracticality of meeting the Code coverage requirements for the subject nozzle-to-vessel welds, and the reasonable assurance provided by the examinations that were completed, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

3. CONCLUSION

The INEEL staff evaluated the licensee's submittal and concluded that certain inservice examinations cannot be performed to the extent required by the Code at the Robert Emmit Ginna Nuclear Plant. For Requests for Relief 1, Rev. 1, and 5 (fourth interval), the licensee's proposed alternative to the Code requirements provides an acceptable level of quality and safety. Therefore, it is recommended that the proposed alternatives be authorized pursuant to 10 CFR 50.55a(a)(3)(i).

For Requests for Relief 42 Revision 1(Part A) (third interval), and 4 (fourth interval), it is concluded that the Code requirements would result in a hardship without a compensating increase in the level of quality and safety. Therefore, it is recommended that these proposed alternatives be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

For Requests for Relief 42 (Part B), 42 (Part C), 43 (third interval), and 3, 6, 7, 8, 14 (Part A), 14 (Part B) and 15 (fourth interval), it is concluded that the Code requirements are impractical for the subject welds. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

For Request for Relief 2 (fourth interval), it is concluded that the licensee is meeting the current Code requirements. Therefore, relief is not required.

Robert Emmit Ginna, Unit 1
Third and Fourth 10-Year ISI Interval

SUMMARY OF RELIEF REQUESTS

Relief Request Number	INEEL TLR Sec.	System or Component	Exam Category	Item No.	Volume or Area to be Examined	Required Method	Licensee Proposed Alternative	Relief Request Disposition
42 Rev. 1 (Part A) Third Interval	2.1	Reactor Pressure Vessel	B-A	B1.11	Lower Shell-to-Ring Forging Weld	Volumetric	Partial volumetric examination be found acceptable	Authorized 10 CFR50.55a(a)(3)(ii)
42 Rev. 1 (Part B) Third Interval	2.2	Reactor Pressure Vessel	B-A	B1.11 B1.30	Circumferential Shell Weld Shell-to-Flange Weld	Volumetric	Partial volumetric examination be found acceptable	Granted 10 CFR50.55a(g)(6)(i)
42 Rev. 1 (Part C) Third Interval	2.3	Reactor Pressure Vessel	B-D	B3.90 B3.100	RPV Nozzle-to-Vessel Weld RPV Nozzle Inside Radius	Volumetric	Partial volumetric examination be found acceptable	Granted 10 CFR50.55a(g)(6)(i)
43 Third Interval	2.4	RHR Heat Exchanger	C-B	C2.32	Nozzle to Shell Weld	Volumetric	Partial volumetric examination be found acceptable	Granted 10 CFR50.55a(g)(6)(i)
1 Rev. 1 Fourth Interval	2.5	Reactor Pressure Vessel	B-A	B1.30 B1.40	Shell-to-Flange Weld Head-to-Flange	Volumetric Volumetric/Surface	Defer Examinations to the End of the Interval	Authorized 10 CFR50.55a(a)(3)(i)
2 Fourth Interval	2.6	Class 1 Valves	B-M-2	B12.50	Valve Body Internal Surfaces	Visual Examination (VT-3)	Perform visual examinations when valves are disassembled for maintenance	Relief not required
3 Fourth Interval	2.7	Class 1 Piping	B-J	B9.11	Terminal End-to-Cast Elbow Circumferential Welds	Volumetric	Perform volumetric examination to maximum extent possible.	Granted 10 CFR50.55a(g)(6)(i)
4 Fourth Interval	2.8	Class 1 Heat Exchangers	B-B B-D	B2.60 B2.80 B3.150	Tubesheet-to-Shell Welds Tubesheet-to-Head Welds Primary Side Nozzle Welds	Volumetric	Perform Volumetric Examinations on only one of three identical RHE	Authorized 10 CFR50.55a(a)(3)(ii)
5 Fourth Interval	2.9	Weld Filler Material	NA	NA	Weld Filler Material	Suitability Evaluation	Utilize Code Cases 2142 and 2143.	Authorized 10 CFR50.55a(a)(3)(i)
6 Fourth Interval	2.10	Class 2 Heat Exchangers	B-D	C3.160	Inner Radius Examinations	Volumetric	None	Granted 10 CFR50.55a(g)(6)(i)
7 Fourth Interval	2.11	Class 2 Piping	C-C	C3.20	Integrally Welded Attachments	Surface	Perform surface examinations to maximum extent possible	Granted 10 CFR50.55a(g)(6)(i)
8 Fourth Interval	2.12	Class 1 Piping	B-F	B5.10	Dissimilar Metal Welds in Vessel Nozzles	Volumetric/Surface	Perform surface examinations to maximum extent possible	Granted 10 CFR50.55a(g)(6)(i)
14 (Part A) Fourth Interval	2.13	Reactor Pressure Vessel	B-A	B1.11 B1.30	Circumferential Shell Weld Shell-to-Flange Weld	Volumetric	Partial volumetric examination be found acceptable	Granted 10 CFR50.55a(g)(6)(i)
14 (Part B) Fourth Interval	2.14	Reactor Pressure Vessel	B-D	B3.90 B3.100	Nozzle-to-Vessel Weld Nozzle Inside Radius	Volumetric	Partial volumetric examination be found acceptable	Granted 10 CFR50.55a(g)(6)(i)
15 Fourth Interval	2.15	RHR Heat Exchanger	C-B	C2.32	Nozzle-to-Shell Weld	Volumetric	Partial volumetric examination be found acceptable	Granted 10 CFR50.55a(g)(6)(i)