CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION
VISSING,G.S.

See Reports

SUBJECT: Forwards fifteen relief requests that will be utilized for Ginna NPP fourth interval ISI program that will start on Jan 1,2000.Attachment 1 includes summaries & detailed description of each relief request.

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ROBERT C. MECREDY
Vice President
Nuclear Operations

October 8, 1999

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I
Washington, D.C. 20555

Subject: Ginna Nuclear Power Plant Inservice Inspection
ASME Section XI Required Examinations, Relief Requests
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

The purpose of this letter is to submit fifteen (15) Relief Requests that will be utilized for our Fourth Interval ISI Program that will start on January 1, 2000. Enclosed as Attachment 1 is a summary of our Relief Requests as well as a detailed description of each individual relief request. Table 1 identifies each relief request, the subject of each relief, and identifies the reference paragraph of 10 CFR 50.55a dealing with the type of relief that is being requested. The first 13 individual relief requests contained within the body of Attachment 1 are virtually identical to those approved for use during our Third Interval Program (minor differences are explained in Table 1 of Attachment 1). A cross reference to the Third Interval Relief Request is identified on the top of each current proposed relief request for your convenience. Relief Requests 14 and 15 are identical to Relief Requests 42 and 43, recently submitted for our Third Interval ISI program.

It should be noted that our Fourth Interval ISI Program has been developed to ASME Section XI, 1995 Edition, 1996 Addenda, as described in our August 26, 1999 submittal of proposed Relief Request 35. RG&E will also implement the EPRI Performance Demonstration Initiative (PDI) to implement mandatory Appendix VIII, consistent with 10 CFR 50.55a (b)(2)(xv) of the final 10 CFR 50.55a Rule effective November 22, 1999. Furthermore, for ASME Section XI repairs and replacements, ASME Section III Code, 1995 Edition, 1996 Addenda shall be used as applicable. Finally, we will be maintaining the current requirements for IWE and IWL, which will be ASME Section XI, 1992 Edition, 1992 Addenda. This is the existing code of record per 10 CFR 50.55a.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

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A047

Attachment

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I
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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
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U.S. NRC Ginna Senior Resident Inspector

TABLE 1
SUMMARY OF RELIEF REQUESTS

RELIEF REQUEST NUMBER	10 CFR RELIEF REFERENCE	RELIEF REQUEST SUBJECT
1	50.55a(a)(3)(ii)	Defer RPV Examinations to End of Interval.
2	50.55a(a)(3)(ii)	Visual Internal Examination of Class 1 valves
3	50.55a(g)(5)(iii)	Category B-J. Cast Stainless Base Metal Ultrasonic Examination.
4	50.55a(a)(3)(ii)	Regenerative Heat Exchanger (RHE), Class 1 Examinations. (Note: The reference to Item Number B3.160 per the 3rd Interval Relief Request Number 18-1 has been moved to Relief Request 6 below.)
5	50.55a(a)(3)(ii)	Alloy 690 Weld Material, Fabrication of New Steam Generators.
6	50.55a(g)(5)(iii)	Regenerative Heat Exchanger (RHE), Inner Radius Examinations.
7	50.55a(g)(5)(iii)	Component Support Integral Attachment Examination Limitations.
8	50.55a(g)(5)(iii)	Surface Examination limitations associated with weld examinations of identified Class 1 safe end nozzles associated with the Reactor Pressure Vessel. (Note: Reflects Surface Examination percentage actually achieved during the 1999 Outage Examination.)
9	50.55a(a)(3)(ii)	Visual Examination of metal containment seals & gaskets.
10	50.55a(a)(3)(ii)	Successive Examinations to repaired areas.

TABLE 1
SUMMARY OF RELIEF REQUESTS

RELIEF REQUEST NUMBER	10 CFR RELIEF REFERENCE	RELIEF REQUEST SUBJECT
11	50.55a(a)(3)(ii)	Bolt torque or tension test for bolted connections.
12	50.55a(a)(3)(i)	VT-2 Visual Examination following repair, replacement, or modification.
13	50.55a(a)(3)(ii)	Minimum illumination and maximum direct examination distance of Class CC components.
14	50.55a(g)(5)(iii)	Reactor Pressure Vessel (RPV) Weld Examination Limitations.
15	50.55a(g)(5)(iii)	Residual Heat Removal (RHR) Heat Exchanger Outlet Nozzle Weld Examination Limitation.

RELIEF REQUEST NO. 1
(Was previously approved as Relief Request No. 1 for Third Interval)
DEFER RPV EXAMINATIONS TO END OF INTERVAL

I. System/Component for Which Relief is Requested:

Relief is requested for the Reactor Pressure Vessel (RPV) Shell-to-Flange and Head-to-Flange Welds.

II Code Requirement

ASME Section XI, 1995 Edition 1996 Addenda, Category B-A, Item Numbers B1.30 and B1.40.

III. Code Requirement from Which Relief is Requested:

Table IWB-2500-1, Examination Category B-A, Item Number B1.30, requires that the RPV Shell-to-Flange weld be examined during the first and third periods in conjunction with the nozzle examinations, with at least 50 percent examined during the first period and the remainder by the end of the third period. The required Shell-to-Flange examination is impractical if performed during the periods specified as it can only be accomplished from the flange surface. When performing this weld examination from the Vessel inside surface, qualified UT techniques can be utilized to perform a more reliable examination with greater examination coverage as demonstrated during our 1999 10-Year Vessel examination.

Table IWB-2500-1, Examination Category B-A, Item Number B1.40, requires that the RPV Head-to-Flange weld be examined from the flange face only during the first and second periods, provided that these same portions are examined from the head during the third period.

IV. Basis for Relief:

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.

RELIEF REQUEST NO. 1 (Con't)

V. Alternate Examinations:

R.E. Ginna Nuclear Power Plant proposes that the examinations identified above be performed from both the flange surface and the vessel wall at or near the end of interval.

VI. Justification for Granting of Relief

During the first three inspections intervals, 100 percent of the accessible length of the RPV welds including the Shell-to-Flange welds were examined at or near the end of the interval when the entire examination could be performed from both the flange surface and the vessel wall. This is a more practical approach in that the required examinations from both surfaces can be performed at the same time. 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. Also, Class 1 Leakage examinations are performed each outage to ensure system/component integrity.

VII. Implementation Schedule:

Implementation shall be performed for the Fourth Interval.

RELIEF REQUEST NO. 2
VISUAL INTERNAL EXAMINATION OF CLASS 1 VALVES
(Was previously approved as Relief Request No. 5 for Third Interval)

I. System/Components for Which Relief is Requested:

Class 1 valves requiring valve body internal VT examination.

Size (In.)	Valve No.	MFG/Type	Line No.
10	842A	Darling/Check	10A-SI2-1502-A
10	842B	Darling/Check	10A-SI2-2501-B
10	867A	Darling/Check	10A-SI2-2501-A
10	867B	Darling/Check	10A-SI2-2501-B
10	700	Velan/Gate	10A-RC02501-A
10	701	Velan/Gate	10A-RC0-2501-A
10	720	Velan/Gate	10A-RC0-2501-B
10	721	Velan/Gate	10A-RC0-2501-B
6	853A	Velan/Check	6A-RC-2501-A
6	853B	Velan/Check	6A-RC-2501-B
6	852A	Velan/Gate	6A-RC-2501-A
6	852B	Velan/Gate	6A-RC-2501-B

II. Code Requirements:

ASME Section XI, 1995 Edition 1996 Addenda, Category B-M-2, Item Number B12.50.

III. Code Requirement from Which Relief is Requested

Table IWB-2500-1, Examination Category B-M-2 Item B12.50, requires an internal VT-3 examination on at least one valve within each group of valves that are of the same size, constructional design (such as globe, gate or check valves) and manufacturing method, that perform similar functions in the system. 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.

RELIEF REQUEST NO. 2 (Con't)

III (Con't)

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.

IV. Basis for Relief:

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.

V. Proposed Alternative Examinations:

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.

VI. Justification for Granting of Relief:

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

RELIEF REQUEST NO. 2 (Con't)

VII. Implementation Schedule:

Implementation shall be performed for the Fourth Interval.

RELIEF REQUEST NO. 3

(Was previously approved as Relief Request no. 17 for Third Interval)

CATEGORY B-J, CAST STAINLESS BASE METAL ULTRASONIC EXAMINATION

I. Components for Which Relief is Requested:

Class 1, Category B-J Item B9.11, Reactor Coolant Cast Pump Terminal End to Cast Elbow Circumferential Welds

Weld ID	Type	% Coverage
PL-FW-XIII	Elbow-to-Pump	45%
PL-FW-XV	Elbow-to-Pump	50%

II. Code Requirements:

ASME Section XI, 1995 Edition, 1996 Addenda, Category B-J, Item Number B9.11.

III. Code Requirements from Which Relief is Requested:

1. Category B-J within Table IWB-2500-1, specifies volumetric examination requirements that are to be performed on welds at Terminal Ends, as well as other circumferential welds associated in the system.

2. Code Case N-460 specifies that a reduction in examination coverage on any Class 1 or Class 2 weld may be accepted provided the reduction in coverage for that weld is less than 10%.

RELIEF REQUEST NO. 3 (Con't)

IV. <u>Basis for Relief:</u>

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 technique 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.

RELIEF REQUEST NO. 3 (Con't)

V. Alternate Examinations:

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.

VI. Justification for Granting of Relief:

Volumetric examination on cast Austenitic and Dissimilar Metal welds may preclude meaningful examinations because of geometry and attenuation variables. 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 due to the high levels of background radiation emitting from these areas. Volumetric examination of the cast stainless welds based on state-of-the-art techniques and associated achievable examination coverage shall be performed.

VII. Implementation Schedule:

Implementation shall be performed for the Fourth Interval.

RELIEF REQUEST NO. 4
(Was previously approved as Relief Request no. 18-1 for Third Interval)
REGENERATIVE HEAT EXCHANGER (RHE), CLASS 1 EXAMINATIONS

I. System/Component for Which Relief is Requested:

Chemical and Volume Control System; Regenerative Heat Exchanger (RHE), Class 1 portion.

II. Code Requirements:

ASME Section XI, 1995 Edition, 1996 Addenda, Category B-B, Item Numbers B2.60, B2.80, and Category B-D, Item Numbers B3.150

III. Code Requirements from Which Relief is Requested:

In accordance with Table IWB-2500-1, the following Categories and Item Numbers require volumetric examination.

Category	Item Number	
B-B	B2.60	Tubesheet-to-Head Welds
B-B	B2.80	Tubesheet-to-Shell Welds
B-D	B3.150	Nozzle-to-Vessel Welds

IV. Basis:

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.

RELIEF REQUEST NO. 4 (Con't)

IV (Con't)

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 the 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.

V. Alternate Examinations:

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.

VI. Justification for Granting Relief:

The RHE consist of three (3) shell and tube heat exchangers connected in series. The bottom heat exchanger operates at a higher temperature of all the units and is the most highly stressed. By limiting the examinations to one heat exchanger will significantly reduce radiation exposure to personnel.

VII. Implementation Schedule:

Implementation shall be performed for the Fourth Interval.

RELIEF REQUEST NO. 5

(Was previously approved as Relief Request No. 27 for Third Interval)

ALLOY 690 WELD MATERIAL, FABRICATION OF NEW STEAM GENERATORS

I. System/Components for Which Relief is Requested:

For the incorporation of Code Cases 2142 and 2143 into the ASME Section XI Program. These Cases allow the use of Alloy 690 weld material.

II. Code Requirements:

ASME Section XI, 1995 Edition with 1996 Addenda, IWA-4224 Requirements.

III. ASME Requirements from which Relief is Requested:

Relief is requested from the requirements specified in IWA-4224 of the 1995 Edition, with 1996 Addenda of the ASME Section XI Code.

> IWA-4224.3 Use of a Different Material
>
> (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.

IV. Basis for Relief:

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.

RELIEF REQUEST NO. 5 (Con't)

IV (Con't)

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.

V. Alternate Examinations:

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".

VI. Justification for Granting of Relief:

Both UNS W86152 and UNS N06052 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 N86182) materials. The new weld materials are the preferred choice for welding applications involving Alloy 690 that was utilized in Fabricating the new Steam Generators.

VII. Implementation Schedule:

Utilization of these materials may be used during the Fourth Interval if and when needed.

RELIEF REQUEST NO. 6

(Was previously approved as Relief Request no. 32 for Third Interval)

REGENERATIVE HEAT EXCHANGER (RHE), INNER RADIUS EXAMINATIONS

I. System/Components for Which Relief is Requested:

Chemical and Volume Control System; Regenerative Heat Exchanger (RHE), Class 1 Inner Radius Examinations;

RHE-N1
RHE-N5

II. Code Requirements:

ASME Section XI, 1995 Edition with 1996 Addenda, Category B-D, Item Number B3.160.

III. Code Requirements from Which Relief is Requested:

In accordance with Table IWB-2500-1, Examination Category B-D, Item Number B3.160.

IV. Basis for Relief:

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.

RELIEF REQUEST NO. 6 (Con't)

IV (Con't)

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.

V. <u>Alternate Examinations:</u>

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.

RELIEF REQUEST NO. 6 (Con't)

VI. Justification for Granting of Relief:

It has been proven that an acceptable Code examination of the RHE nozzle inner radius region is not possible utilizing current technology. It demonstrates that signals from different depth notches located in the most optimum position of the mock-up could not be differentiated from noise and geometric reflectors without the aid of finger damping on the ID surface. The evaluations have also shown that the limitation in the boss area, in the 0 and 180 degree position, along with the transition to these areas, limits the size of transducers that can be employed and where the transducers can be placed.

VII. Implementation Schedule:

Implementation shall be performed for the Fourth Interval.

References:

1. EPRI Report Title " Evaluation of Ultrasonic Examination Technology for Inspection of Regenerative Heat Exchanger Nozzles at the Rochester Gas & Electric, Ginna Nuclear Plant" Dated 17 April 1996 by Douglas E. MacDonald and E. Kim Kietzman.

2. AEA Reactor Services Report Title "Report on the Mathematical Modeling of Two Nozzle Specimens for Rochester Gas & Electric Corporation" Dated 16 November 1992 by P. D. Birchall and L. N. J. Poulter.

RELIEF REQUEST NO. 7
(Was previously approved as Relief Request No. 34 for Third Interval)
COMPONENT SUPPORT INTEGRAL ATTACHMENT EXAMINATION LIMITATIONS

I. System/Component(s) for which Relief is Requested:

This Relief Request is requested for three (3) supports. Inspections of these supports is addressed by Class 2, Category C-C, Item Number C3.20, Identified Component Support Integral Attachments Surface Examinations.

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

II. Code Requirement:

ASME Section XI Code, 1995 Edition, 1996 Addenda, requires essentially 100% of the weld length to obtain code coverage. ASME Section XI Code Case N-460 states that if the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in coverage is acceptable provided that the coverage (the lack of) is less than 10%.

III. Code Requirement from which Relief is Requested:

Relief is requested from examining 100% of the length for these 3 supports. Examining 100% of the weld length would be impractical, because the support would have to be redesigned or replaced to enable inspection. The amount of coverage is estimated at 50% for MSU-33, 50% for MSU-34, and 82% for Penetration 140.

RELIEF REQUEST NO. 7 (Con't)

IV. Basis for Relief:

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 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.

V. Alternate Examinations:

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.

RELIEF REQUEST NO. 7 (Con't)

VI. Justification for the Granting of Relief:

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 made accessible for future NDE examinations. Due to the original limited design accessibility, ISI examination coverage cannot be obtained to the extent required by the current ASME Code.

ASME Section XI 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. The identified examination coverage for these items should be acceptable in fulfilling ASME Section XI coverage requirements.

VII. Implementation Schedule:

Implementation shall be performed for the Fourth Interval.

RELIEF REQUEST NO. 8

(Was previously approved as Relief Request No. 36 for Third Interval)

REACTOR PRESSURE VESSEL NOZZLE-to-SAFE END BUTT WELD SURFACE EXAMINATION LIMITATIONS

I. System/Component(s) for which Relief is Requested:

This Relief Request is requested for six (6) Reactor Pressure Vessel Nozzle-to-Safe End Butt Welds. Inspection of these welds is addressed under Class 1, Category B-F, Item Number B5.10, Nozzle-to-Safe End Weld Surface Examinations as identified below.

Weld ID #	ISI Summary #	Coverage Obtained
PL-FW-II	002100	62%
PL-FW-V	002400	75%
PL-FW-IV	002700	68.5%
PL-FW-VII	003000	72%
AC-1003-1	003300	0%(*)
AC-1002-1	003600	0%(*)

Note: (*) = welds embedded in concrete.

II. Code Requirement:

ASME Section XI Code, 1995 Edition, 1996 Addenda, Category B-F, Item Number B5.10, volumetric and surface examinations shall be performed with essentially 100% of the weld length to obtain code coverage. ASME Section XI Code Case N-460 states that if the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in coverage is acceptable provided that the (lack of) coverage is less than 10%. Previous Codes utilized did not include this 90% coverage requirement and examinations were performed to the extent obtainable.

RELIEF REQUEST NO. 8 (Cont'd)

III. Code Requirement from which Relief is Requested:

Relief is requested from the surface examination requirements for the six (6) identified welds. Surface Examination of the first four (4) welds is limited due to Original Construction Code interferences of the floor and wall in the "Sandbox" where these welds are located. The "Sandboxes" would have to be redesigned to enable the welds to be surface examined to obtain Code required coverage. (Volumetric examination of these welds is performed from the inside of the Vessel and is not a part of this Relief Request.)

Surface Examination of the last two welds is impractical. The concrete surrounding the Reactor Pressure Vessel has embedded these welds. The concrete wall around the Reactor Pressure Vessel would have to be redesigned or replaced to enable the two (2) welds to be inspected with a surface examination. (Volumetric examination of these welds is performed from the inside of the Vessel and is not a part of this Relief Request.)

IV. Basis for Relief:

Relief is requested pursuant to the provisions of 10 CFR 50.55a(g)(5)(iii), the required examination coverage for the identified welds is impractical and would require redesign or replacement to obtain Code required surface examination coverage.

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.

RELIEF REQUEST NO. 8 (Cont'd)

IV (Con't)

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.

V. Alternate Examinations:

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.

VI. Justification for the Granting of Relief:

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 made accessible for future NDE examinations. Due to the original limited design accessibility or lack of design accessibility, ISI surface examination coverage can not be obtained to the extent required by the ASME Code. 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. The identified examination coverage for these items should be acceptable in fulfilling ASME Section XI coverage requirements.

RELIEF REQUEST NO. 8 (Cont'd)

VII. Implementation Schedule:

The surface examinations will be performed during the end of the Fourth Interval on the first four (4) accessible welds. For the two (2) welds embedded in concrete, surface examinations can not be performed.

RELIEF REQUEST NO. 9
(Was Relief Request No, 37 for Third Interval)
CONTAINMENT INSPECTION SEALS AND GASKETS

I. System/Component(s) for which Relief is Requested:

Seals and gaskets of Class MC pressure retaining components and metallic liners of Class CC components, Examination Category E-D, Item Numbers E5.10 and E5.20 of IWE-2500, Table IWE-2500-1, ASME Section XI, 1992 Edition, 1992 Addenda. Several hundred seals and gaskets are affected by this relief request.

II. Code Requirement:

ASME Section XI Code, 1992 Edition, 1992 Addenda, IWE-2500, Table IWE-2500-1, Category E-D, Item Numbers E5.10 and E5.20 requires seals and gaskets on airlocks, hatches, and other devices to be visually examined, VT-3, once each interval to assure containment leak-tight integrity.

III. Code Requirement from which Relief is Requested:

Relief is requested from performing the Code-required visual examination, VT-3, on the above identified metal containment seals and gaskets.

IV. Basis for Relief:

10 CFR 50.55a was amended in the Federal Register (61 FR 41303) to require the use of the 1992 Edition, 1992 Addenda, of Section XI when performing containment examinations. The penetrations discussed below contain seals and gaskets.

RELIEF REQUEST NO. 9 (Cont'd)

A. Electrical Penetrations

Electrical penetrations use a header plate attached to a containment penetration nozzle flange with redundant O-rings between the header plate and flange face. Modules through which electrical conductors pass are installed in the header plate. One type, manufactured by Amphenol, uses seals and gaskets to assure leak tight integrity. A second type, manufactured by Conax, uses a set of compression fittings. Replacement modules for the Amphenol penetrations use a combination of O-rings and compression fittings. Each penetration is pressurized with dry nitrogen to maintain and monitor integrity and to prevent the intrusion of moisture into the penetration. These seals and gaskets cannot be inspected without disassembly of the penetration to gain access to the seals and gaskets.

B. Containment Personnel and Equipment Hatches

The Personnel and Equipment Hatches utilize an inner and outer door with gasket surfaces to ensure a leak tight integrity. These hatches also contain other gaskets and seals such as the handwheel shaft seals, electrical penetrations, blank flanges, and equalizing pressure connections which require disassembly to gain access to the gaskets and seals.

Seal and gasket joints receive a 10 CFR 50 Appendix J test. As noted in 10 CFR 50 Appendix J, the purpose of Type B tests is to measure leakage of containment or penetrations whose design incorporates resilient seals, gaskets, sealant compounds, and electrical penetrations fitted with flexible metal seal assemblies. Examination of seals and gaskets require the joints, which are proven adequate through Appendix J testing, be disassembled. For electrical penetrations, this would involve a pre-maintenance Appendix J test, de-termination of cables at electrical penetrations if enough cable slack is not available, disassembly of the joint, removal and examination of the seals and gaskets, reassembly of the joint, re-termination of the cables if necessary, post maintenance testing of the cables, and a post maintenance Appendix J test of the penetration. The work required for the Containment Hatches would be similar except for the de-termination, re-termination, and testing of cables. This imposes the risk that equipment could be damaged. The 1992 Edition, 1993 Addenda, of Section XI recognizes that disassembly of joints to perform these examinations is not warranted. Note 1 in Examination Category E-D was modified in the 1995 Edition of Section XI to state that seal or gasket connections need not be disassembled solely for performance of examinations. However, without disassembly, most of the surface of the seals and gaskets would be inaccessible.

RELIEF REQUEST NO. 9 (Cont'd)

IV (Con't)

For those penetrations that are routinely disassembled, a Type B test is required upon final assembly and prior to start-up. Since the Type B test will assure the leak tight integrity of primary containment, the performance of the visual examination would not provide an increase in the level of safety or quality.

Seals and gaskets are not part of the containment pressure boundary under current Code rules (NE-1220 (b)). The airlocks and hatches containing these materials are tested in accordance with 10 CFR 50, Appendix J. If increased leakage is identified during these Appendix J tests, the cause of leakage would be investigated. If increased leakage were due to degradation of the seal or gasket material, corrective measures would be applied and the component retested. Repair or replacement of seals and gaskets is not subject to Code (1992 Edition, 1992 Addenda) rules in accordance with Paragraph IWA-4111(b)(5) of ASME Section XI.

The visual examination of seals and gaskets in accordance with IWE-2500, Table IWE-2500-1 is a burden without any compensating increase in the level of safety or quality.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii). Compliance with the original requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Testing the seal and gasket joints in accordance with 10 CFR 50, Appendix J will provide adequate assurance of the leak-tight integrity of the seals and gaskets.

RELIEF REQUEST NO. 9 (Cont'd)

V. Alternate Examinations:

The leak testing of seal and gasket joints will be in accordance with 10 CFR 50, Appendix J. No additional alternative examinations to the visual examination, VT-3, of the seals and gaskets will be performed.

VI. Justification for the Granting of Relief:

This Relief Request is similar to Relief Request E-1 submitted by Davis-Besse as one of the EPRI "Containment Inspection Program Guide" Pilot Plant Relief Requests. This Relief Request will minimize Ginna operating and maintenance cost without decreasing the level of quality and safety.

VII. Implementation Schedule:

Relief is requested for the first inspection interval for the IWE Containment Inspection Program (1996 - 2008). Note that this interval overlaps the Third and Fourth 10-Year Interval inspections of the Ginna Inservice Inspection Program.

RELIEF REQUEST NO. 10
(Was Relief Request No. 38 for Third Interval)
CONTAINMENT INSPECTION SUCCESSIVE EXAMINATIONS AFTER REPAIR

I. System/Component(s) for which Relief is Requested:

All Class MC, Paragraphs IWE-2420 (b) and IWE-2420(c) successive examination requirements for components found acceptable for continued service.

II. Code Requirement:

Paragraphs IWE-2420 (b) and IWE-2420(c) of the 1992 Edition, 1992 Addenda of ASME Section XI requires that when component examination results require evaluation of flaws, evaluation of areas of degradation, or repairs in accordance with Article IWE-3000, and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period listed in the schedule of the inspection program of Paragraph IWE-2411 or Paragraph IWE-2412, in accordance with Table IWE-2500-1, Examination Category E-C.

III. Code Requirement from which Relief is Requested:

Relief is requested from the requirements of ASME Section XI Code, 1992 Edition, 1992 Addenda, Paragraphs IWE-2420 (b) and IWE-2420(c) to perform successive examination on repairs.

IV. Basis for Relief:

10 CFR 50.55a was amended in the Federal Register (61 FR 41303) to require the use of the 1992 Edition, 1992 Addenda, of Section XI when performing containment examinations. The purpose of a repair is to restore the component to an acceptable condition for continued service in accordance with the acceptance standards of Article IWE-3000. Paragraph IWA-4150 requires the owner to conduct an evaluation of the suitability of the repair including consideration of the cause of failure.

RELIEF REQUEST NO. 10 (Cont'd)

IV (Con't)

If the repair has restored the component to an acceptable condition, it is overly conservative to require successive examinations. Other paragraphs of the ASME Code recognize this requirement as overly conservative. If the repair was not suitable, then the repair does not meet code requirements and the component is not acceptable for continued service. Neither Paragraph IWB-2420 (b), Paragraph IWC-2420(b), nor Paragraph IWD-2420 (b) requires a repair to be subject to successive examination requirements. Furthermore, if the repair area is subject to accelerated degradation, it would still require augmented examination in accordance with Table IWE-2500-1, Examination Category E-C. The successive examination of repairs in accordance with Paragraphs IWE-2420 (b) and IWE-2420(c) constitutes a burden without a compensating increase in quality or safety.

In their resolution to public comment # 3.3, the NRC stated within SECY 96-080 dated April 17, 1996: "The purpose of IWE-2420 (b) is to manage components found to be acceptable for continued service (meaning no repair or replacement at this time) as an Examination Category E-C component. If the component had been repaired or replaced, then the more frequent examination would not be needed."

Relief is requested in accordance with 10 CFR 50.55a(a)(3) (ii). Compliance with the original requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The requirement to perform successive examinations following repairs has been removed in the 1997 Addenda of ASME Section XI. This addenda has been approved by the ASME Main Committee and has been published.

V. <u>Alternate Examinations:</u>

Successive examinations in accordance with Paragraphs IWE-2420 (b) and IWE-2420(c) are not required for repairs made in accordance with Article IWA-4000.

RELIEF REQUEST NO. 10 (Cont'd)

VI. Justification for the Granting of Relief:

This Relief Request is similar to Relief Request E-6 submitted by Davis-Besse as one of the EPRI "Containment Inspection Program Guide" Pilot Plant Relief Requests. This Relief Request will minimize Ginna operating and maintenance cost without decreasing the level of quality and safety.

VII. Implementation Schedule:

Relief is requested for the first inspection interval for the IWE Containment Inspection Program (1996 - 2008). Note that this interval overlaps the Third and Fourth 10-Year Interval inspections of the Ginna Inservice Inspection Program.

RELIEF REQUEST NO. 11
(Was Relief Request No. 39 for Third Interval)
CONTAINMENT INSPECTION BOLT TORQUE OR TENSION TESTING

I. System/Component(s) for which Relief is Requested:

Class MC pressure retaining bolting.

II. Code Requirement:

ASME Section XI, 1992 Edition with the 1992 Addenda, Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item E8.20. Bolt torque or tension testing is required on bolted connections that have not been disassembled and reassembled during the inspection interval.

III. Code Requirement from which Relief is Requested:

Relief is requested from ASME Section XI 1992 Edition with the 1992 Addenda Table IWE-2500-1 Examination Category E-G, Pressure Retaining Bolting, Item E8.20. Bolt torque or tension testing is required on bolted connections that have not been disassembled and reassembled during the inspection interval.

IV. Basis for Relief:

10CFR50.55a was amended in the Federal Register (61 FR 41303) to require the use of the 1992 Edition, 1992 Addenda, of Section XI when performing containment examinations. Bolt torque or tension testing is required on bolted connections that have not been disassembled and reassembled during the inspection interval.

RELIEF REQUEST NO. 11 (Cont'd)

IV (Con't)

Determination of the torque or tension value would require that the bolting be un-torqued and then re-torqued or re-tensioned. The performance of the Type B test itself proves that the bolt torque or tension remains adequate to provide a leak rate that is within acceptable limits. The torque or tension value of bolting only becomes an issue if the leak rate is excessive. Once a bolt is torqued or tensioned, it is not subject to dynamic loading that could cause it to experience significant change.

Verification of torque or tension values on bolted joints that are proven adequate through Appendix J testing and visual inspection is adequate to demonstrate that design function is met. Torque or tension testing is not required on any other ASME Section XI; Class 1, 2, or 3 bolted connections or their supports as part of

the Inservice inspection program. Also, all penetrations at R.E. Ginna Nuclear Power Plant are seated with pressure (not unseated).

Relief is requested in accordance with 10 CFR 50.55a(a)(3) (ii). Compliance with the original requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

V. <u>Alternate Examinations:</u>

The following examinations and tests required by Subsection IWE ensure the structural integrity and the leak-tightness of Class MC pressure retaining bolting, and, therefore, no additional alternative examinations are proposed:

RELIEF REQUEST NO. 11 (Cont'd)

V (Con't)

(1) Exposed surfaces of bolted connections shall be visually examined in accordance with requirements of Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item No. E8.10, and

(2) Bolted connections shall meet the pressure test requirements of Table IWE-2500-1, Examination Category E-P, All Pressure Retaining Components, Item E9.40, and

(3) A general visual examination of the entire containment once each inspection period shall be conducted in accordance with 10 CFR 50.55a(b)(2)(x)(E).

VI. Justification for the Granting of Relief:

This Relief Request is similar to Relief Request E-7 submitted by Davis-Besse as one of the EPRI "Containment Inspection Program Guide" Pilot Plant Relief Requests. This Relief Request will minimize Ginna operating and maintenance cost without decreasing the level of quality and safety.

VII. Implementation Schedule:

Relief is requested for the first inspection interval for the IWE Containment Inspection Program (1996 - 2008). Note that this interval overlaps the Third and Fourth 10-Year Interval inspections of the Ginna Inservice Inspection Program.

RELIEF REQUEST NO. 12
(Was Relief Request No. 40 for Third Interval)
CONTAINMENT INSPECTION VT-2 AFTER REPAIR, REPLACEMENT OR MODIFICATION

I. System/Component(s) for which Relief is Requested:

All components subject to the rules and requirements for repair, replacement or modification of Class MC, IWE-5000 system pressure testing visual examination in accordance with the 1992 Edition, 1992 Addenda of ASME Section XI.

II. Code Requirement:

Paragraph IWE-5240 of the 1992 Edition, 1992 Addenda of ASME Section XI requires that the requirements of Paragraph IWA-5240 for visual examination, VT-2, are applicable following repair, replacement or modification.

III. Code Requirement from which Relief is Requested:

Relief is requested from performing the VT-2 visual examination in connection with system pressure testing following repair, replacement or modification under Article IWE-5000.

IV. Basis for Relief:

10 CFR 50.55a was amended in the Federal Register (61 FR 41303) to require the use of the 1992 Edition, 1992 Addenda, of Section XI when performing containment examinations. Paragraph IWE-5210 states that except as noted within Paragraph IWE-5240, the requirements of Article IWA-5000 are not applicable to Class MC or Class CC components.

RELIEF REQUEST NO. 12 (Cont'd)

IV (Con't)

Paragraph IWE-5240 states that the requirements of Paragraph IWA-5240 (corrected from IWA-5246 to IWA-5240 in the 1993 Addenda) for visual examinations are applicable. Paragraph IWA-5240 identifies a "VT-2" visual examination. VT-2 examinations are conducted to detect evidence of leakage from pressure-retaining components, with or without leakage collection systems, as required during the conduct of a system pressure test. In addition, personnel performing VT-2 examinations are required to be qualified in accordance with Subarticle IWA-2300 of ASME Section XI.

For repairs, replacements or modifications that are performed under ASME Section XI Code, applicable Construction Code (or Installation Code) non-destructive examinations (NDE) are performed and must meet the acceptance criteria of the Construction / Installation Code. In addition to the Construction Code NDE, applicable ASME Section XI NDE pre-service NDE are also performed. These Construction Code and Section XI pre-service NDE requirements provide additional assurances that the repairs, replacements or modifications are sound and leak-tight. Table IWE-2500-1, Examination Category E-P, identifies the examination method of 10 CFR 50, Appendix J and does not specifically identify a VT-2 visual examination. 10 CFR 50, Appendix J provides requirements for testing as well as acceptable leakage criteria. These tests are performed by Appendix J "Test" personnel and utilize calibrated equipment to determine acceptability. Additionally, 10 CFR 50.55a(b)(2)(x)(E) requires a general visual examination of the containment each period that would identify any structural degradation that may contribute to leakage. A "VT-2" visual examination will not provide additional assurance of safety beyond that of current Appendix J practices.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). The proposed alternative (pressure testing in accordance with 10 CFR 50, Appendix J) will provide an acceptable level of quality and safety.

The reference to paragraph IWA-5240 has been removed in the 1997 Addenda of ASME Section XI. This addenda has been approved by the ASME Main Committee and should be published in 1998.

RELIEF REQUEST NO. 12 (Cont'd)

V. Alternate Examinations:

Testing shall be conducted in accordance with 10 CFR 50, Appendix J, in lieu of Paragraph IWE-5240 of ASME Section XI, as well as performing an IWE detailed visual examination (VT-1) of the required or replaced area.

VI. Justification for the Granting of Relief:

For repairs, replacements or modifications that are performed under ASME Section XI Code, applicable Construction Code (or Installation Code) non-destructive examinations (NDE) are performed and must meet the acceptance criteria of the Construction / Installation Code. In addition to the Construction Code NDE, applicable ASME Section XI NDE pre-service NDE are also performed. These Construction Code and Section XI pre-service NDE requirements provide additional assurances that the repairs, replacements or modifications are sound and leak-tight. This Relief Request will minimize Ginna operating and maintenance cost without decreasing the level of quality and safety.

VII. Implementation Schedule:

Relief is requested for the first inspection interval for the IWE Containment Inspection Program (1996 - 2008). Note that this interval overlaps the Third and Fourth 10-Year Interval inspections of the Ginna Inservice Inspection Program.

RELIEF REQUEST NO. 13
(Was Relief Request No. 41 for Third Interval)
CONTAINMENT INSPECTION REMOTE VT OF CLASS CC

I. System/Component(s) for which Relief is Requested:

All components subject to the rules and requirements for Inservice Inspection of Class CC Concrete Components, Examination Category L-A, Concrete, Item L.1.11 as applicable to IWL-2310, Visual Examination and Personnel Qualification and IWA-2210, Visual Examinations.

II. Code Requirement:

ASME Section XI, 1992 Edition, 1992 Addenda, IWL-2310, Visual Examination and Personnel Qualification and IWA-2210, Visual Examinations requires specific minimum illumination and maximum direct examination distance for all concrete surfaces.

III. Code Requirement from which Relief is Requested:

Relief is requested from Paragraph IWA-2210, Visual Examination Requirements for minimum illumination and maximum direct examination distance of Class CC components under Paragraph IWL-2310.

RELIEF REQUEST NO. 13 (Cont'd)

IV. Basis for Relief:

10 CFR 50.55a was amended in the Federal Register (61 FR 41303) to require the use of the 1992 Edition, 1992 Addenda, of Section XI when performing containment examinations. In addition to the requirements of Subsection IWL, the Rulemaking also imposes the requirements of Subsection IWA of the 1992 Edition, 1992 Addendum, of ASME Section XI for minimum illumination and maximum direct examination distance of Class CC components, specifically for the examination of concrete under Paragraph IWL-2510. When remotely performing the visual examinations required by Subsection IWL Paragraph IWL-2510, the maximum direct examination distance specified in Table IWA-2210-1 may be extended, and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased. IWA-2210 allows for remote examination as long as the remote examination procedure is demonstrated to resolve the selected test chart characters. The Registered Professional Engineer (RPE) will identify minimum size of indications of interest. For remote visual examination, the procedure and equipment to be used will be demonstrated capable of resolving these minimum indications to the satisfaction of the RPE and the Authorized Nuclear Insurance Inspector (ANII), as allowed in IWA-2240, "Alternative Examinations." The record of demonstration will be available to Regulatory Authorities.

RELIEF REQUEST NO. 13 (Cont'd)

IV (Con't)

Accessibility to higher portions of the dome and the containment building itself make it a hardship to obtain the maximum direct examination distance and minimum illumination requirements. The installation of extensive temporary scaffold systems or a climbing scaffold system to access these portions of the containment would be necessary. These scaffolds would provide limited access due to containment geometry restrictions as well as structural and equipment interferences. The installation and removal of these scaffolds would increase both worker radiation exposure and personnel safety in order to meet Paragraph IWA-2210 requirements. The NRC staff received seven comments that were consolidated into Public Comment # 2.3 in Part III of Attachment 6A to SECY-96-080. The Staff response to these concerns is as follows: "Comments received from ASME members on the containment committees indicate that the newer, more stringent requirements of IWA-2210 were not intended to be used for the examination of containments and were inadvertently included in Subsection IWL. The NRC agrees that remote examinations are the only practical method for inspecting much of the containment surface area. 50.55a(b)(2) (x)(B) has been added to the final rule which contains alternative lighting and resolution requirements which may be used in lieu of the requirements contained in IWA-2210-1." However, as specified within 10CFR50.55a(b)(2)(x)(B) of the final rule, this alternative applies only to Subsection IWE, and not to Subsection IWL.

Relief is requested in accordance with 10 CFR 50.55a (a)(3) (ii). Compliance with the original requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

V. <u>Alternate Examinations:</u>

When performing remotely the visual examinations required by Subsection IWL, Paragraph IWL-2510, the maximum direct examination distance specified in Table IWA-2210-1 may be extended, and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

RELIEF REQUEST NO. 13 (Cont'd)

VI. Justification for the Granting of Relief:

This Relief Request is similar to Relief Request L-1 submitted by Calvert Cliffs as one of the EPRI "Containment Inspection Program Guide" Pilot Plant Relief Requests. Relief Request L-1 (for Calvert Cliffs) was subsequently approved by the NRC. Refer to a letter from the NRC (S. Bajwa) to Baltimore Gas and Electric (C. Cruse), dated November 16, 1998 (Docket No. 50-317/318). This Relief Request will minimize Ginna operating and maintenance cost without decreasing the level of quality and safety.

VII. Implementation Schedule:

Relief is requested for the first inspection interval for the IWL Containment Inspection Program (1996 - 2008). Note that this interval overlaps the Third and Fourth 10-Year Interval inspections of the Ginna Inservice Inspection Program.

RELIEF REQUEST NO. 14
(Was Relief Request No. 42 for Third Interval)
REACTOR PRESSURE VESSEL (RPV) WELD EXAMINATION LIMITATIONS

I. System/Component(s) for which Relief is Requested:

This Relief Request pertains to eight (8) Reactor Pressure Vessel welds or Inner Radius Volumetric examinations. Volumetric examination limitations of these eight welds are identified in Attachment Number 1.

II. Code Requirement:

When performing Volumetric examinations, ASME Section XI Code, 1995 Edition, 1996 Addenda, requires essentially 100% of the weld length or area to obtain coverage.

ASME Section XI Code Case N-460 states that if the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in coverage is acceptable provided that the coverage (the lack of) is less than 10%.

III. Requirement from which Relief is Requested:

Relief is requested from examining 100% of the weld length or areas for these eight (8) identified items. Examining 100% of the weld length or areas would be impractical due to original design interference. Attachment Number 1 identifies volumetric examination achievable coverage and associated limitations.

RELIEF REQUEST NO. 14 (Cont'd)

IV. Basis for Relief:

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) items identified within Attachment Number 1 was 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.

V. Alternate Examinations:

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.

RELIEF REQUEST NO. 14 (Cont'd)

VI. Justification for the Granting of Relief:

The Reactor Pressure Vessel was designed and constructed to ASME Section III, 1965 Edition construction code. This code did not contain requirements to ensure that items be made accessible for future NDE examinations. Due to the original limited design accessibility, examination coverage can not be obtained to the extent required by the ASME Section XI Code.

ASME Section XI Class 1 leakage examinations are performed each outage as well as Operator walkdowns as specified by Plant Operating Procedures. These operator walkdowns and 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.

Previous examinations were performed on these items in conformance to the Code requirements in effect for RG&E at those times.

VII. Implementation Schedule:

Implementation shall be performed at the end of the Fourth Interval.

RELIEF REQUEST NO. 14 (Cont'd)

Attachment Number 1

Category Number	Item Number	Sum. Number	Weld ID	Description/Requirement	%	Limitations
B-A	B1.30	000501 000502 000503	RPV-A	Vessel to Flange Circ. Weld Section XI Code Required	54%	Keyways & Irradiation Slots
B-A	B1.11	000300	RPV-D	Lower Shell to Ring Forging Circ. Weld. Section XI Code Required	81%	Guide Lugs & Incores
B-D	B3.90	001900	N1A	Nozzle Vessel WD 028D-30M. Section XI Code Required	55%[1] 70%[2]	Nozzle Boss
B-D	B3.90	002500	N1B	Nozzle Vessel WD 208D-30M. Section XI Code Required	55%[1] 70%[2]	Nozzle Boss
B-D	B3.100	002300	N2A -IRS	Nozzle Inside Radius Section. Section XI Code Required	90%	Inner Radius
B-D	B3.100	002900	N2B -IRS	Nozzle Inside Radius Section. Section XI Code Required	90%	Inner Radius
B-D	B3.90	003100	AC- 1003	Nozzle Vessel WD 108D-30M. Section XI Code Required	55%[1] 72%[2]	Nozzle Boss
B-D	B3.90	003400	AC- 1002	Nozzle Vessel WD 288D-30M. Section XI Code Required	55%[1] 72%[2]	Nozzle Boss

1 Volumetric Weld Examination

2 Volumetric Near Surface Examination

Request for Relief No. 15
(Was Relief Request No. 43 for Third Interval)
Residual Heat Removal (RHR) Heat Exchanger Outlet Nozzle Welds Examination Limitations

I. System/Component(s) for Which Relief is Requested:

This Relief Request pertains to two (2) Residual Heat Removal (RHR) Heat Exchanger Outlet Nozzle to Shell Welds, ASME Class 2, Category C-B, Item C2.32. There is one nozzle of this type associated with each of the two identical RHR Heat Exchangers. The Code requires that the examination be performed when the component is opened. Both Heat Exchangers were opened and Volumetrically examined. Limited coverage is identified below.

Summary #	Weld ID	Exam Coverage	Limitations
169253	ONSRHE-1B	79%	due to internal welded separation plate
169053	ONSRHE-1A	79%	due to internal welded separation plate

II. ASME Section XI Code Requirement:

ASME Section XI Code, 1995 Edition, 1996 Addenda, requires essentially 100% of the weld length or area to obtain coverage. ASME Section XI Code Case N-460 states that if the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in coverage is acceptable provided that the coverage (the lack of) is less than 10%.

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III. Requirement from Which Relief is Requested:

Relief is requested from examining 100% of the weld length or areas for these two (2) identified items. Examining 100% of the weld length or areas would be impractical due to original design interference.

IV. Basis for Relief:

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 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.

V. Alternate Examinations:

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

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VI. Justification for the Granting of Relief:

The Residual Heat Removal (RHR) Heat Exchangers were designed and constructed to ASME Section VIII, 1965 Edition construction code. This code did not contain requirements to ensure that items be made accessible for future NDE examinations. Due to the original limited design accessibility, examination coverage can not be obtained to the extent required by the current ASME Code.

ASME Section XI periodic leakage examinations are performed as well as Operator walkdowns as specified by Plant Operating Procedures. These operator walkdowns 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.

VII. Implementation Schedule:

These examinations will be performed in accordance with Table IWC-2500-1 for the Fourth Interval.