

December 16, 1996



Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Document Control Desk

SUBJECT: Second Ten Year Inservice Inspection Program

Byron Nuclear Power Station, Units 1 and 2
Facility Operating Licenses NPF-37 and NPF-66
NRC Docket Nos. 50-454 and 50-455

REFERENCE: (1) Letter from M. Lesniak, Commonwealth Edison (ComEd) to Document Control Desk, Nuclear Regulatory Commission (NRC) dated February 23, 1996, transmitting the Byron Station Second Ten Year Inservice Inspection Program.

(2) Letter from George F. Dick, Jr. (NRC), to D.L. Farrar (ComEd) dated June 13, 1996, transmitting a Request For Additional Information (RAI) regarding the Byron Second 10 Year Inservice Inspection Program.

(3) Letter from M. Lesniak, (ComEd) to Document Control Desk, (NRC) dated August 12, 1996, transmitting the Byron Station Responses to Letter in Reference (5) regarding the Second 10 Year Inservice Inspection Program.

(4) Letter from George F. Dick, Jr. (NRC), to Irene Johnson (ComEd) dated October 16, 1996, transmitting a supplementary Request For Additional Information (RAI) regarding the Byron Second 10 Year Inservice Inspection Program.

(5) Letter from George F. Dick, Jr. (NRC), to Irene Johnson (ComEd) dated December 13, 1996, transmitting a supplementary Request For Additional Information (RAI) regarding the Byron Second 10 Year Inservice Inspection Program.

In Reference (1), ComEd submitted the Byron Station Second Ten Year Inservice (ISI) Program and associated relief requests. In Reference (2) the NRC issued a Request for Additional Information concerning the remaining relief requests. ComEd provided a response to that request in Reference (3).

In Reference (4), NRC requested additional information regarding various aspects of the Second 10 Year Inservice Inspection Program (Program Plan) submitted by Byron Station in Reference (5). The answers to the specific questions raised in the RAI are contained in Attachment 1.

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In Reference (5), NRC requested additional information regarding various aspects of the Second 10 Year Inservice Inspection Program (Program Plan) submitted by Byron Station in Reference (1). The answers to the specific questions raised in the RAI are contained in Attachment 2.

In conjunction with the responses to References (4) and (5), ComEd has revised specific relief requests related to the second 10 year interval. The revised relief requests are included in Attachment (3). The attached revised pages are intended to replace the original pages previously provided. All changes have been identified by a revision bar in the right-hand margin, as well as noting the revision on the affected relief request.

Please address any comments or questions regarding this matter to this office.

Sincerely,



Marcia T. Lesniak
Nuclear Licensing Administrator

attachments

cc: A. B. Beach, NRC Regional Administrator, RIII
G. F. Dick, Jr., Byron Project manager, NRR
S. Burgess, Senior Resident Inspector, Byron
Office of Nuclear Safety, IDNS
Michael T. Anderson, INEL

ATTACHMENT 1
RAI QUESTIONS AND RESPONSES

1. Unapproved Code Cases: In accordance with 10CFR50.55a(c)(3), 10CFR50.55a(d)(2), and 10CFR50.55a(e)(2), American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) cases may be used as alternatives to the Code requirements. Code cases that the NRC has approved for use are listed in Regulatory Guide (RG) 1.147, *Inservice Inspection Code Case Acceptability*, with any additional conditions deemed necessary by the NRC. Code cases not referenced in RG 1.147 may be adopted only if authorized by the NRC on a case-by-case basis. In some cases, use of unapproved Code cases may be acceptable for use when certain conditions are included. To ensure consistent implementation, licensees proposing the use of currently unapproved Code case(s), must commit to such conditions, if applicable.

In Relief I2R-10, the licensee has proposed to implement the alternatives contained in ASME Code Case N-522, *Pressure Testing of Containment Penetration Piping*. This Code case may be considered acceptable for use with the following conditions:

- a) The leak test is performed at peak calculated containment design pressure; and
- b) a test procedure is used that provides for detect and location of through wall leakages in pipe segments that are being tested.

In Request for Relief I2R-17, the use of Code Case N-509, *Alternate Rules for Selection and Examination of Class 1, 2, and 3 Integrally Welded Attachments*, is proposed. This Code case may be considered acceptable for use provided that a minimum of 10 percent of the total number of integral attachments in all Class 1, 2, and 3 systems are examined.

To find the proposed alternatives to the Code requirements acceptable for use, incorporation of the above conditions into the applicable request for relief is necessary. Confirm that the conditions stated above will be met.

RESPONSE:

Regarding Request for Relief I2R-10, the leak tests are performed at no less than the peak calculated containment pressure (presently 44.4 psig) as defined in the present Byron Technical Specifications and implementing procedures. This value represents the peak calculated pressure resulting from the design basis accident (with initial pressure of 15.7 psia, per UFSAR Table 6.2-1). The containment design pressure is 50 psig.

Byron Station will implement station VT surveillances to provide for the detection and location of external piping leakage. The results and subsequent actions will be documented through the ISI Pressure Test Program.

Relief Request I2R-17 is revised to include a statement regarding selection of integral attachments under the "Proposed Alternate Examination" section as follows. "The selection of the integral attachments reflect 10% of the total of all nonexempt (per IWX-1220) ASME Class 1, 2, and 3 piping, pump, and valve integral attachments, and in the case of multiple vessels of similar design, function, and service, one integrally welded attachment of only one of the multiple vessels." A copy of the revised relief request is attached.

2. In Request for Relief I2R-03 relief is requested from performing the Code-required volumetric examination of the pressurizer surge nozzle-to-vessel weld and inside radius section. This request was evaluated and authorized for the first 10-year interval provided that the Code-required examination was performed if the insulation was removed for any reason. To find this proposed alternative to the Code requirements acceptable for use for the second 10-year interval, incorporation of the above condition into the request for relief is necessary. Confirm that the condition stated above will be met.

RESPONSE:

As identified in the SER dated May 3, 1996, the pressurizer surge Nozzle-to-vessel weld and nozzle inside radius section will be volumetrically examined if the lower head insulation is removed for any reason. Relief Request I2R-03 has been revised to include this condition. A copy of the revised relief request is attached.

3. Request for Relief I2R-05: In lieu of performing the 100 percent volumetric examination as required by the Code, the licensee requested to perform a "best effort" ultrasonic examination on the residual heat removal heat exchanger nozzle-to-shell welds. In the staff's request for Additional Information dated June 13, 1996, the licensee was requested to provide an estimate of the Code-required volume that could be examined with the best effort ultrasonic examination. This information was not provided in the August 12, 1996 response. Provide an estimate of the coverage that can be achieved with the best effort ultrasonic examination.

As a result of indications found in the subject welds during the 1st 10-year interval, the licensee submitted a request for relief (December 12 1995) from the successive examination requirements of the Code. This request, which was subsequently evaluated in the staff's Safety Evaluation dated February 29, 1996, included the performance of a surface examination once each period as the proposed alternative. Will the welds included in I2R-05 continue to receive surface examinations during the 2nd 10-year interval?

RESPONSE:

The "best effort" ultrasonic examination will primarily be performed from the nozzle outside surface. Anticipated obstructions include the reinforcement fillet weld located directly above the nozzle-to-vessel weld. This fillet weld restricts inspection transducer movement and limits available examination angles. A 70° transducer will be used to pass sound under the fillet weld to reach the lower 1/3T of the pressure retaining weld. Scanning from the other side of the weld will be accomplished by use of a 45° shear wave transducer on the shell surface to reach the examination volume. These scans are in the axial direction of the nozzle-to-vessel weld. Scans in the circumferential directions will not be performed due to the location of the fillet weld. This weld location does not allow the transducer to scan over the examination volume from the outside of the nozzle. It is estimated that 99.92% of the examination volume will be reached in the axial direction. Without the circumferential scans, the total scanning percent is estimated at 56.61%.

Consistent with the SER for the 1st interval dated February 29, 1996, a dye penetrant (surface) examination will be performed on each nozzle in each period for the second ten year interval.

I2R-05 has been revised to include the above conditions. A copy of the revised relief request is attached.

4. In Request for Relief I2R-01, relief is requested from 2 separate requirements for reactor vessel shell Weld RPVC-WR29 due to physical obstructions that limit the volumetric examination to less than 90 percent of the required volume - (1) the periodic volumetric examination required by Section XI for the 2nd interval, and (2) the augmented volumetric examination required by 10CFR50.55a(g)(6)(ii) that should have been performed during the first 10-year interval.

Regarding the augmented examination, the regulations state that licensees that cannot completely satisfy the examination requirements of 10CFR50.55a(g)(6)(ii)(A), must submit information supporting that determination and must propose an alternative that would provide an acceptable level of quality and safety. This alternative may be used when authorized by the NRC staff. In accordance with 10CFR50.55a(a)(3), alternatives to the requirements of 10CFR50.55a(g)(6)(ii)(A) may be used when authorized by the NRC staff if the licensee demonstrates that either (i) the proposed alternative provides an acceptable level of quality and safety, or (ii) that the examination requirements would result in hardship without a compensating increase in quality and safety.

Request for relief I2R-01 can not be evaluated for the Section XI requirements in the 2nd 10-year interval until the augmented volumetric examination requirements of the regulations are satisfied for the 1st interval. To satisfy the regulations, the licensee should provide a separate submittal containing the required proposed alternative. The staff will evaluate the proposed alternative to verify that examination coverage has been maximized from both the vessel interior and exterior. To help the staff complete the evaluation and close out this issue, provide a technical discussion describing how examination coverage was maximized (including the possibility of, or the burden associated with, performing an examination from the vessel exterior). Once the augmented volumetric examination requirements are satisfied for the first interval, the limited Code examinations for the 2nd interval can be addressed by revising and resubmitting Request for Relief I2R-01.

RESPONSE:

With respect to the 1st Interval inspection requirements, Byron Station submitted Unit 1 inspection results via Byron Letter # Byron-96-0202/Relief Request NR-20 on July 15, 1996 to the Document Control Desk. This request for relief detailed the coverages achieved, physical obstructions, and the proposed alternatives for the 1st Interval examination of the Byron Unit 1 reactor vessel. Relief Request NR-20 seeks relief for Byron Unit 2 based on the coverages achieved during the Unit 1 examination. A copy of the July 15, 1996 letter and Relief Request NR-20 is attached for reference.

The information contained within the 2nd Interval Relief Request I2R-01, Revision 1, corresponds to the information contained within the 1st Interval Relief Request NR-20.

Both relief requests I2R-01, Revision 1, and NR-20 describe the physical limitations of the reactor vessel geometry with respect to examination coverages of circumferential shell weld RPVC-WR29 (shell course to Dutchman weld) and RPVC-WR-16 (lower disk to Dutchman weld) from the vessel interior.

Examination from the vessel exterior is obstructed by the concrete structure surrounding the reactor vessel(s). This reinforced concrete structure is the reactor vessel support structure. The vessel is suspended by 4 nozzle supports which distribute the loads through the surrounding concrete structure. The annulus between the vessel in the vicinity of the obstructed welds and the structure is approximately 6". No exterior access to these welds is possible from either above or below the vessel. This concrete structure is the reactor vessel support structure and modifications to allow access to the vessel exterior are not practicable.

ATTACHMENT 2
RAI QUESTIONS AND RESPONSES

1. Relief Requests I2R-01 and I2R-02 were submitted pursuant to 10 CFR 50.55a(a)(3)(ii). However, the bases for these requests mention "physical obstructions and geometric limitations" and "examinations performed to the extent practical." Similarly, Relief Request I2R-05 was submitted pursuant to 10 CFR 50.55a(a)(3)(i), but the alternative is to perform the Code-required examination to the "maximum extent practical." It is unclear as to which paragraph of 10 CFR 50.55a these requests should be evaluated against. Provide appropriate references to the *Code of Federal Regulations* and clarify the supporting bases for the subject requests.

RESPONSE:

Regarding Relief Request I2R-05, the request was revised in response to the NRC RAI dated October 16, 1996, incorporating surface exams as alternate examinations. The appropriate Code of Federal Regulations (CFR) is 10 CFR 50.55a(a)(3)(i) on the basis that the proposed alternative examination provides an acceptable level of quality and safety. A copy of revised relief request I2R-05 is attached.

Regarding Relief Request I2R-01, the request is submitted under the basis that compliance with applicable Code requirements can only be accomplished by redesigning and refabricating the Reactor Vessel and concrete support structure. 10 CFR 50.55a(g)(5)(iii) is cited in a revised request as the appropriate CFF reference, given that modifications to the reactor vessel or support structure are impractical. A copy of revised relief request I2R-01 is attached.

Regarding Relief Request I2R-02, compliance with the applicable Code requirements is considered impractical since full Code compliance can only be accomplished by redesigning and refabricating the Reactor Vessel(s) and/or building a structure surrounding the vessel(s). 10 CFR 50.55a(g)(5)(iii) is cited in a revised request as the appropriate CFR reference, given that modifications to the reactor vessel or support structure are impractical. A copy of revised relief request I2R-02 is attached.

2. Relief Request I2R-17 was submitted without reference to a section of the *Code of Federal Regulations*. Without this reference, the request can not be evaluated. Provide the appropriate reference to the *Code of Federal Regulations*.

RESPONSE:

Relief Request I2R-17 has been revised to incorporate the appropriate reference to 10 CFR 50.55a(a)(3)(i) on the basis that the proposed alternatives of Code Case N-509 provide an acceptable level of quality and safety. A copy of revised relief request I2R-17 is attached.

ATTACHMENT 3
REVISED RELIEF REQUESTS

**BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL**

RELIEF REQUEST I2R-01 Revision 2

(Page 1 of 6)

COMPONENT IDENTIFICATION:

Code Class: 1
Reference: IWB-2500, Table IWB-2500-1
Examination Category: B-A
Item Numbers: B1.11 and B1.21
Description: Limited Volumetric Examination of Reactor Vessel Circumferential Shell Welds
Component Numbers: RPVC-WR16, RPVC-WR29 (Same for both units)
Drawing Numbers: IRPV-1-ISI (Unit 1) and 2RPV-1-ISI (Unit 2)

CODE REQUIREMENT:

Table IWB-2500-1, Examination Category B-A, Item Numbers B1.11 and B1.21 require a 100% volumetric examination of the Reactor Vessel Circumferential Shell welds as detailed in Figures IWB-2500-1 and IWB-2500-3.

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and pursuant to 10CFR50.55a(g)(5)(iii), conformance with these Code requirements is impractical as conformance would require extensive structural modifications to the reactor vessel support structure.

HISTORY:

Byron Station, during Refuel Outage B1R07, conducted ultrasonic examinations of the Byron Unit 1 Reactor Vessel. This was the last refuel outage of the third period of the first Inservice Inspection Interval and occurred in April through June 1996. Framatome Technologies Inc. (FTI) was contracted to perform the examinations with their state-of-the-art "URSULA" manipulator and their "ACCUSONEX" UT system. The examinations were performed in accordance with the requirements in ASME Section XI, Article IWA-2232, USNRC Regulatory Guide 1.150 and 10CFR50.55 a (g)(6)(ii)(A).

Previously granted relief request NR-1 for Byron Station Units 1 and 2 First Inspection Interval for the Reactor Shell welds was subsequently revoked in 10CFR 50.55 a(g)(6)(ii)(A) with respect to examination coverages. During the performance of the B1R07 examinations, physical obstructions and geometry prevented UT coverage in excess of 90% of the required volume for the above listed component numbers. Full 100% UT coverage was obtained for the reactor circumferential shell welds WR-7, WR-18, and WR-34 (Byron Station Reactor Vessel(s) do not have any longitudinal shell welds). The limited amount of examination coverage attained for shell welds WR-16 and WR-29 is provided below in the Basis for Relief.

FTI is contracted to perform the 10 year Reactor Examinations on Byron Unit 2 during the late Fall of 1997 and, as the two Byron reactors are identical, similar coverage percentages are expected.

**BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL**

RELIEF REQUEST I2R-01 Revision 2

(Page 2 of 6)

A. REACTOR VESSEL CIRCUMFERENTIAL SHELL WELDS

The examination of the Unit 1 Lower Shell Course-to-Dutchman weld, RPVC-WR29, is restricted by six (6) core barrel locating lugs welded to the inner surface of the vessel approximately 4 inches above the weld (See Figure 1).

These lugs obstruct the automated UT inspection tool from examining the code required volume of the weld below each lug (156°). The FTI "URSULA" tool has a 6 degree movement arm and the physical size of the lugs and the "yaw" joint of the tool prevented scanning below the lugs back into the weld and surrounding base metal. All weld metal can be examined from both sides where access is available between the lugs (204°). Examinations for perpendicular and parallel reflectors covered areas accounting for 57% of the weld metal and heat affected zone (HAZ). Similarly, 57% of the weld metal can be examined for transverse reflectors from two opposing directions.

The examination of the Unit 2 Lower Shell Course-to-Dutchman weld, RPVC-WR29, is similarly obstructed (see Figure 1). All weld metal can be examined from both sides where access is available between the lugs (204°). Examinations for perpendicular and parallel reflectors can cover areas accounting for 57% of the weld metal and heat affected zone (HAZ). Similarly, 57% of the weld metal can be examined for transverse reflectors from two opposing directions.

B. REACTOR VESSEL LOWER HEAD CIRCUMFERENTIAL WELDS

The examination of the Unit 1 Lower Disk-to-Dutchman weld, RPVC-WR16, is restricted by the 58 instrument tubes that penetrate the lower disk and physically obstruct the UT search unit and/or the search unit position device as shown in Figure 2 and 3. Approximately 19% of the weld length cannot be examined. The weld and the HAZ received essentially 100% coverage for parallel reflectors from the Dutchman side and for transverse reflectors in two opposing directions. Partial coverage is achieved for parallel reflectors from the disk side on the remainder of the weld resulting in an aggregate of all scan coverage of approximately 81%.

The examination of the Unit 2 Lower Disk-to-Dutchman weld, RPVC-WR16, is similarly restricted, see Figure 2 and 3. The weld and the HAZ will receive essentially 100% coverage for parallel reflectors from the Dutchman side and for transverse reflectors in two opposing directions. Partial coverage can be achieved for parallel reflectors from the disk side on the remainder of the weld resulting in an aggregate of all scan coverage of approximately 81%.

For welds referenced in Sections A and B above, the probability of a flaw occurring only in one of the areas not being examined is extremely small. Most future indications of significant size will be found by the examination of the weld as it is currently performed.

In addition, a VT-2 examination during system pressure testing per Category B-P is also performed on the Reactor Vessel each refueling outage to verify leaktight integrity of these welds.

Examination from the vessel exterior is obstructed by the concrete structure surrounding the reactor vessel(s). This reinforced concrete structure is the reactor vessel support structure. The vessel is suspended by 4 nozzle supports which distribute the loads through the surrounding concrete structure. The annulus between the vessel in the vicinity of the obstructed welds and the structure is approximately 6". No exterior access to these welds is possible from either above or below the vessel. This concrete structure is the reactor vessel support structure and modifications to allow access to the vessel exterior are not practicable.

Compliance with the applicable Code requirements can only be accomplished by redesigning and refabricating the Reactor Vessel and concrete support structure. Byron Station deems this course of action a hardship without a compensating increase in the level of quality and safety.

**BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL**

RELIEF REQUEST I2R-01 Revision 2

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PROPOSED ALTERNATE EXAMINATION

The ultrasonic examination of the Byron Unit 1 reactor vessel was performed to the maximum extent possible. No alternative volumetric examination is proposed to examine the areas not scanned due to obstructions or geometric constraints.

VT-1 inspection was conducted on the weld(s) and HAZ(s) from the inside clad surface utilizing a submersible robot during the Byron Unit 1 Refuel Outage B1R07. Additionally, a VT-2 examination during system pressure testing per Category B-P is performed on the Reactor Vessel each refueling outage to verify leaktight integrity of these welds.

The ultrasonic examination of the Byron Unit 2 reactor vessel will also be performed to the maximum extent possible. The same obstructions and geometric constraints are expected to limit the examination. The visual examinations will be repeated for the Unit 2 RPV welds.

JUSTIFICATION:

The Code required volumetric examination has been completed to the maximum extent practical using ultrasonic examination techniques for Byron Unit 1. The RPV examinations are conducted using an automated technique from the I.D. of the vessel. Access to allow inspection from the O.D. (shell side) of these welds is restricted due to the structural concrete surrounding the vessel.

Reasonable assurance of the continued inservice structural integrity of the subject welds is achieved without performing a complete Code examination. The weld(s) have received visual examinations (VT-1 and VT-2) to visually verify the integrity of the welds.

Compliance with the applicable Code requirements can only be accomplished by redesigning and refabricating the Reactor Vessel(s) and/or building a structure surrounding the vessel(s). Byron Station believes this course of action is a hardship without a compensating increase in the level of quality and safety.

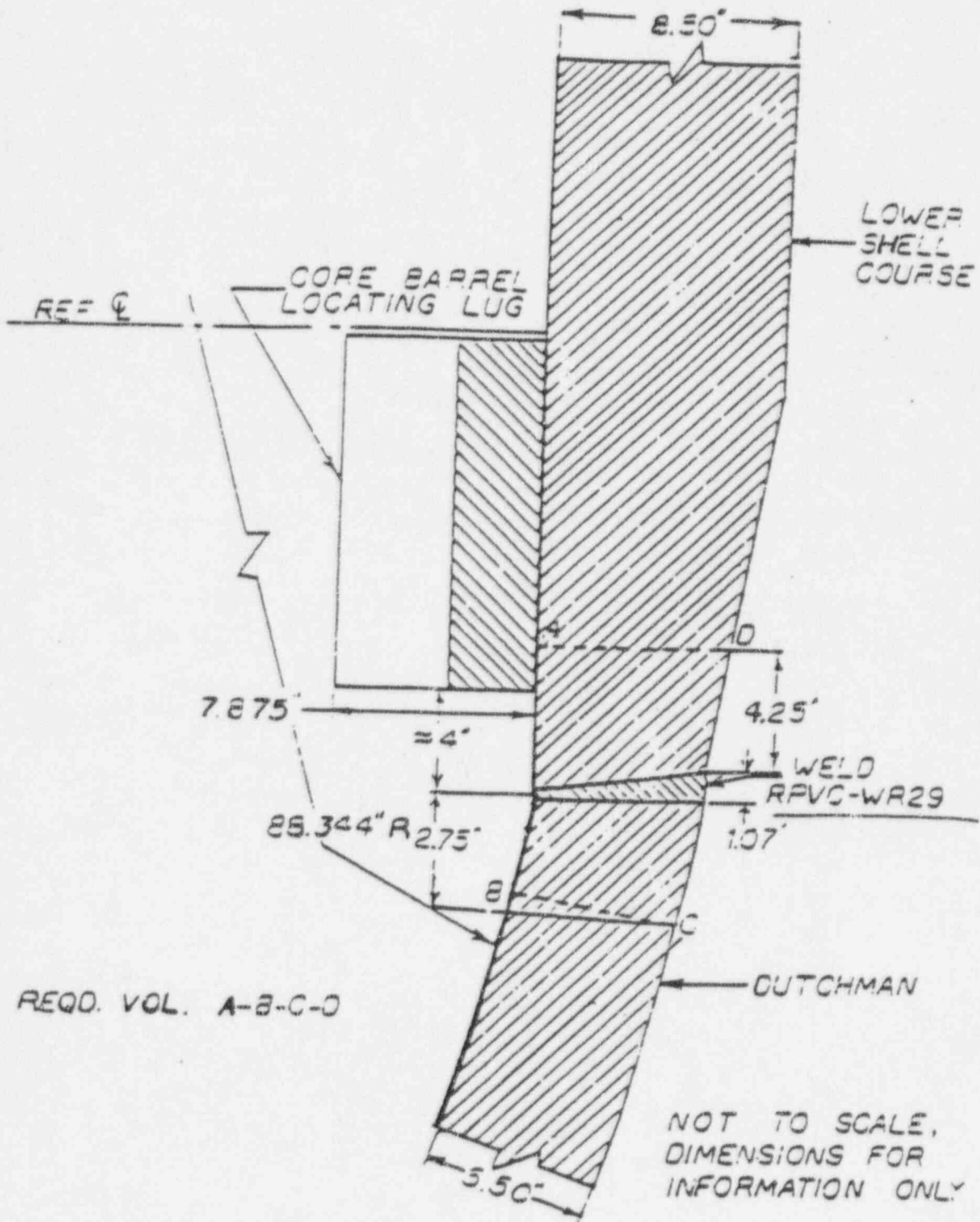
PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the second inspection interval for Byron Units 1 and 2.

BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL

RELIEF REQUEST I2R-01 Revision 2
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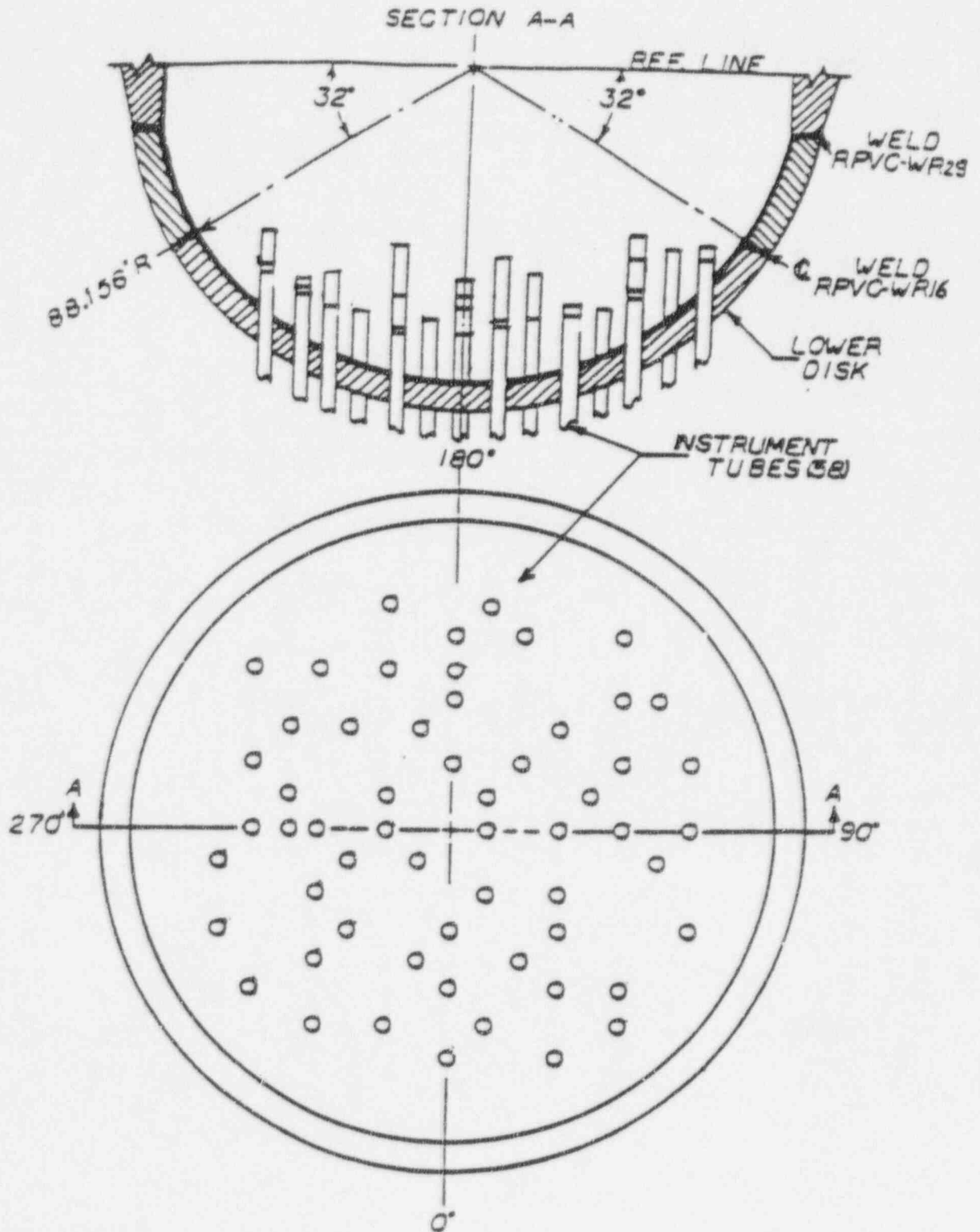
FIGURE 1



BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL

RELIEF REQUEST I2R-01 Revision 2
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FIGURE 2

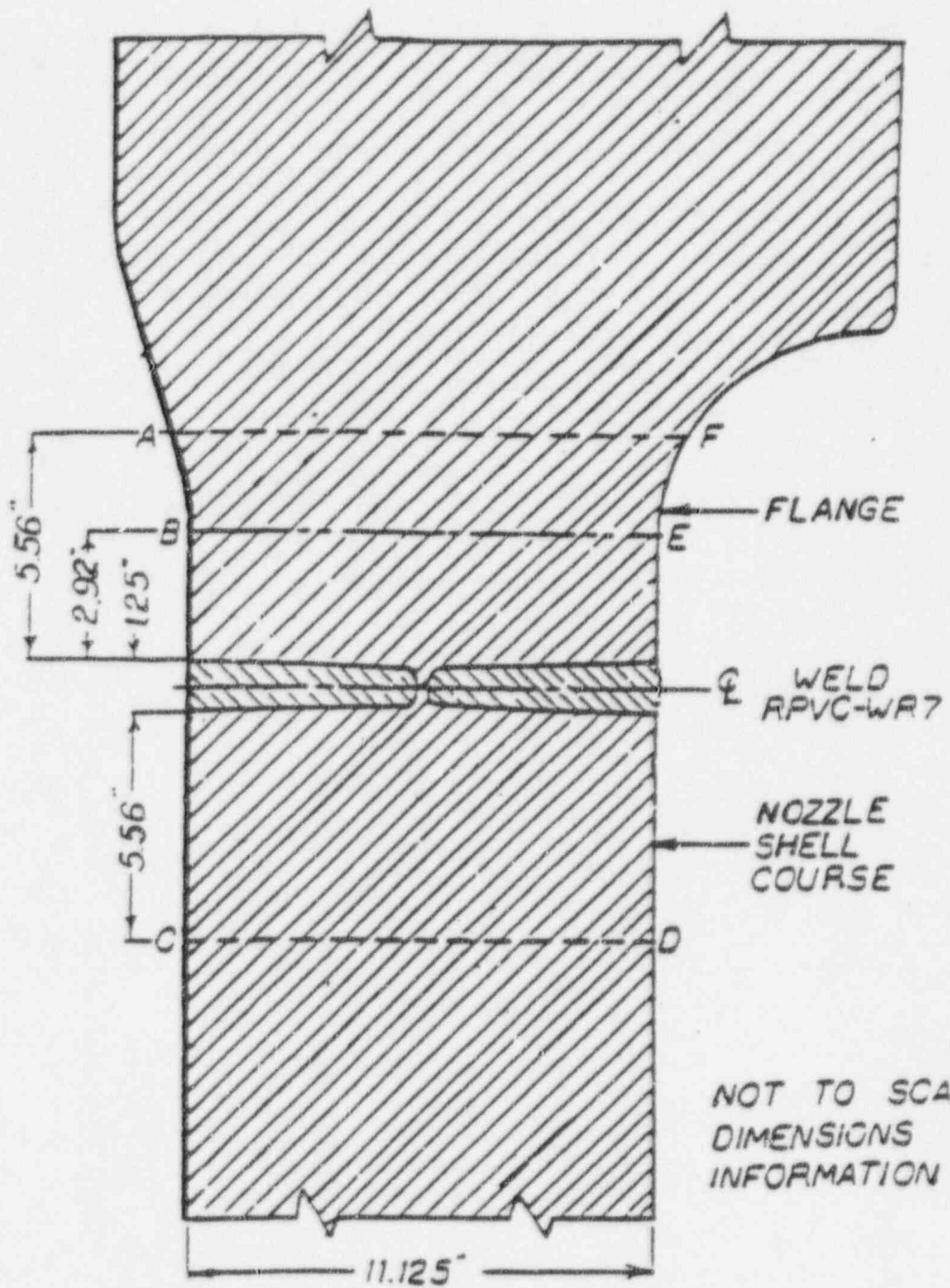


BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL

RELIEF REQUEST I2R-01 Revision 2

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FIGURE 3



NOT TO SCALE,
DIMENSIONS FOR
INFORMATION ONLY

REQUIRED VOLUME A-C-D-F
VOLUME MISSED A-B-E-F

**BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL**

RELIEF REQUEST I2R-02 Revision 2

(Page 1 of 3)

COMPONENT IDENTIFICATION

Code Class: 1
References: IWB-2500, Table IWB-2500-1
Examination Category: B-D
Item Number: B3.90
Description: Limited Volumetric Examination of the Reactor Vessel Outlet Nozzle to-Vessel Welds
Component Numbers: RPVN-A, RPVN-D, RPVN-E, and RPVN-H
(Same weld numbers both units)
Drawing Number: 1RPV-1-ISI (Unit 1) and 2RPV-1-ISI (Unit 2)

CODE REQUIREMENT

Table IWB-2500-1, Examination Category B-D, Item Number B3.90 requires a 100% volumetric examination of the Reactor Vessel Nozzle-to-Vessel Welds as detailed in Figure IWB-2500-7.

HISTORY:

Byron Station, during Refuel Outage B1R07, conducted ultrasonic examinations of the Byron Unit 1 Reactor Vessel. This was the last refuel outage of the third period of the first Inservice Inspection Interval and occurred in April through June 1996. Framatome Technologies Inc. (FTI) was contracted to perform the examinations with their state-of-the-art "URSULA" manipulator and their "ACCUSONEX" UT system. The examinations were performed in accordance with the requirements in ASME Section XI, Article IWA-2232, USNRC Regulatory Guide 1.150 and 10CFR50.55 a (g)(6)(ii)(A). The nozzle-to-shell welds were scanned from the shell for reflectors oriented perpendicular to the weld axis. Scans were performed from the nozzle bore using a 0° longitudinal wave and 45° shear wave transducers looking for reflectors oriented parallel to the weld axis.

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested on the basis that conformance with the Code requirements is impractical as conformance would require extensive structural modifications to the reactor vessel support structure and surrounding concrete structure.

The Reactor Vessel Outlet Nozzles were designed and fabricated with an extension lip as shown in Figure 1 such that the nozzle-to-vessel welds are partially obstructed for the code required volumetric examination. This obstruction limits the circumferential scan for reflectors transverse to the weld such that 19% (aggregate) of the required volume can not be examined.

Strict ASME Section III quality controls were used when designing, fabricating, and installing these welds. In addition, these welds were volumetrically examined during Preservice Inspections (PSI) with no irregularities found. The probability of a flaw occurring only in one of the areas not being examined is extremely small. Most future indications of significant size will be found by the examination of the weld as it currently exists.

Based on the above information, reasonable assurance of the continued inservice structural integrity of the subject welds is achieved without performing a complete the Code examination. In addition, a VT-2 examination during system pressure testing per Category B-P is also performed on the Reactor Vessel each refueling outage to verify leaktight integrity of these welds.

**BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL**

RELIEF REQUEST I2R-02 Revision 2

(Page 2 of 3)

JUSTIFICATION:

The Code required volumetric examination has been completed to the maximum extent practical using ultrasonic examination techniques for Byron Unit 1. The RPV examinations are conducted using an automated technique from the I.D. of the vessel. Examination from the vessel exterior is obstructed by the reinforced concrete structure surrounding the reactor vessel(s) and the reactor vessel nozzle supports. The vessel is suspended by 4 nozzle supports which distribute the loads through the surrounding concrete structure. The annulus between the vessel, in the vicinity of the nozzle to vessel welds, and the structure is approximately 6" or less. No exterior access to these nozzle to vessel welds is possible from either above or below the nozzles. This concrete structure is the reactor vessel support structure and modifications to allow access to the nozzles exterior are not practicable.

VT-1 inspection was conducted on the weld(s) and HAZ(s) from the inside clad surface utilizing a submersible robot during the Byron Unit 1 Refuel Outage B1R07. Additionally, a VT-2 examination during system pressure testing per Category B-P is performed on the Reactor Vessel each refueling outage to verify leaktight integrity of these welds.

The ultrasonic examination of the Byron Unit 2 reactor vessel will also be performed to the maximum extent possible. The same obstructions and geometric constraints are expected to limit the examination. The visual examinations will be repeated for the Unit 2 RPV welds.

Compliance with the applicable Code requirements can only be accomplished by redesigning and refabricating the Reactor Vessel(s) and/or building a structure surrounding the vessel(s). Byron Station believes this course of action is a hardship without a compensating increase in the level of quality and safety.

PROPOSED ALTERNATE EXAMINATION

None; the Code required volumetric examination will be completed to the maximum extent practical using ultrasonic examination techniques.

FTI is contracted to perform the 10 year Reactor Examinations on Byron Unit 2 during the late Fall of 1997 and, as the two Byron reactors are identical, similar coverage percentages are expected.

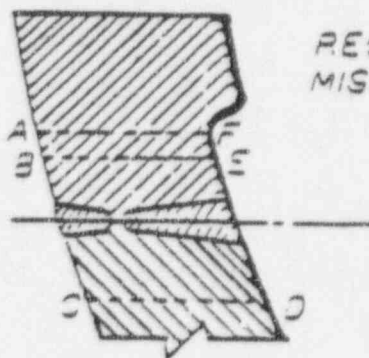
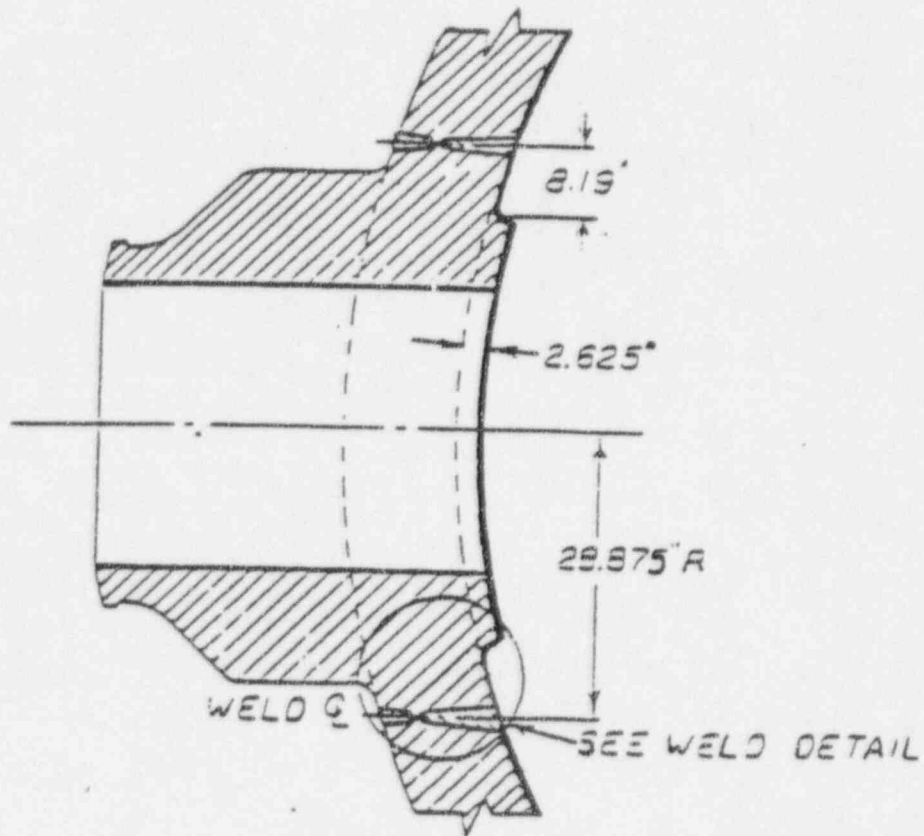
PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the second inspection interval.

BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL

RELIEF REQUEST I2R-02 Revision 2
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Figure 1



REQD. VOLUME A-C-D-F
MISSED VOLUME A-B-E-F

NOT TO SCALE.
DIMENSIONS FOR
INFORMATION ONLY

**BYRON STATION UNITS 1&2 SECOND INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL**

RELIEF REQUEST I2R-03 Revision 2

(Page 1 of 5)

COMPONENT IDENTIFICATION

Code Class: 1
References: IWB-2500, Table IWB-2500-1
Examination Category: B-D
Item Number: B3.110 and B3.120
Description: Limited Volumetric Examination of the Pressurizer Surge Nozzle-to-Vessel Head Weld and Surge Nozzle Inside Radius Section
Component Numbers: 1RY-01-S, PN-01 and PN-01-NIR (Unit 1)
2RY-01-S, PN-01 and PN-01-NIR (Unit 2)
Drawing Numbers: 1PZR-1-ISI (Unit 1) and 2PZR-1-ISI (Unit 2)

CODE REQUIREMENT

Table IWB-2500-1, Examination Category B-D, Item Numbers B3.110 and B3.120 require a 100% volumetric examination of Pressurizer Nozzle-to-Vessel Welds and Pressurizer Nozzle Inner Radius Section as detailed in Figure IWB-2500-7(b).

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The Byron Unit 1 and 2 Pressurizers include a single surge nozzle which is welded to the lower head as shown in Figure 1. In order to perform the code required volumetric examinations on the nozzle-to-vessel weld and the nozzle inside radius section, the outer surface of the lower vessel head must be accessible for proper surface preparation and ultrasonic scanning.

The lower head is normally covered by a 4 inch thick, multi-layered stainless steel insulation which was not designed for removal. In order to remove the insulation, the 78 heater penetration cables shown in Figure 2 would have to be disconnected. In addition, each of the 78 convection stops, which are riveted to the insulation would have to be cut to facilitate the insulation removal per Figure 3.

The radiation exposure to plant personnel for the insulation removal, surface preparation, and examination is estimated to be 150 person rem, based on an area dose rate of .5 R/hour.

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Even with the insulation removed, full volumetric examination coverage of the nozzle-to-vessel weld can not be achieved. The surge nozzle geometry limits ultrasonic transducer contact, and thus scanning on the nozzle side of the weld is impractical. On the vessel side of the weld, the heater penetrations obstruct scanning such that only a small percentage of the weld volume could be captured.

Very limited volumetric examination of the nozzle inner radius section is achievable from the outside surface of the pressurizer with the insulation removed. The blend region would not be accessible to allow for an adequate surface preparation and examination. A limited exam would be possible if scanning was performed from the nozzle side; however, due to the complex geometry of the nozzle, the resulting coverage would provide very limited data from which to assess the condition of the inner radius.

Volumetric examination of the nozzle-to-head weld and nozzle inner radius section is also not practical from the vessel inside surface. The inside surface is accessible only by removing the manway. The radiation exposure for the removal and reinstallation of the manway is estimated to be approximately 2 person-rem. In addition, the internal baffle plates would obstruct access to the debris screen and surrounding inside surfaces of the nozzle, thus prohibiting a meaningful visual VT-1 examination.

Based on the above information, the code required volumetric examination of the pressurizer nozzle-to-vessel lower head weld and associated nozzle inner radius section is deemed impractical. Even partial compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety-the personnel radiation hazards associated with limited data obtained by partial volumetric examination is not justified. Reasonable assurance of the continued inservice structural integrity of the subject welds can be achieved by performing a VT-2 examination during system pressure testing per Category B-P.

Compliance with the applicable Code requirements can only be accomplished by redesigning and refabricating the Pressurizer. Byron Station deems this course of action a hardship without a compensating increase in the level of quality and safety.

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PROPOSED ALTERNATE EXAMINATION

None. The Code required VT-2 examination during system pressure testing per Category B-P is performed on the Pressurizer each refueling outage to verify leaktight integrity of these areas.

As identified in the SER dated May 3, 1996, the pressurizer surge nozzle-to-vessel weld and nozzle inside radius section will be volumetrically examined if the lower head insulation is removed for any reason.

PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the second inspection interval.

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FIGURE 1:
Pressurizer Lower Head Assembly

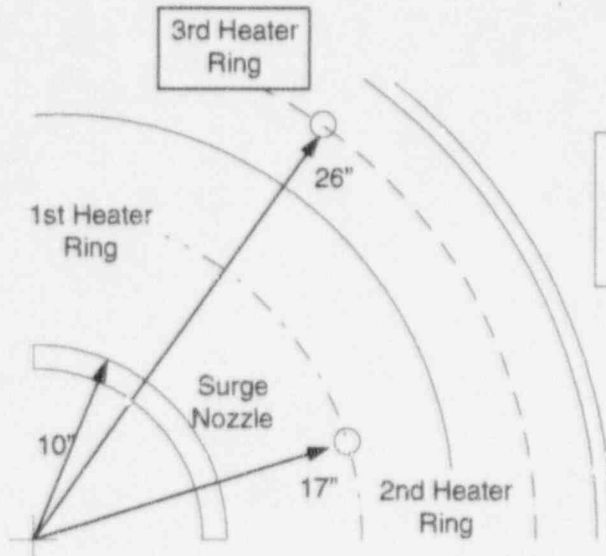
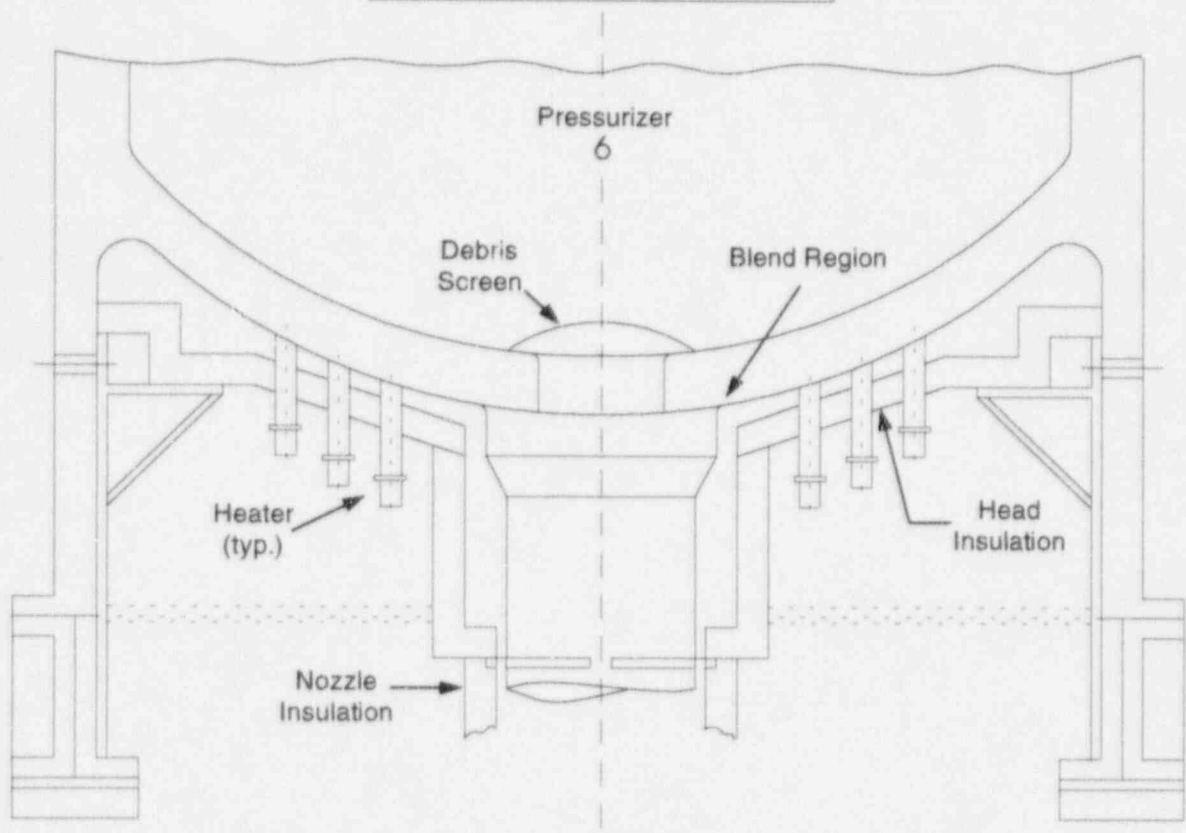
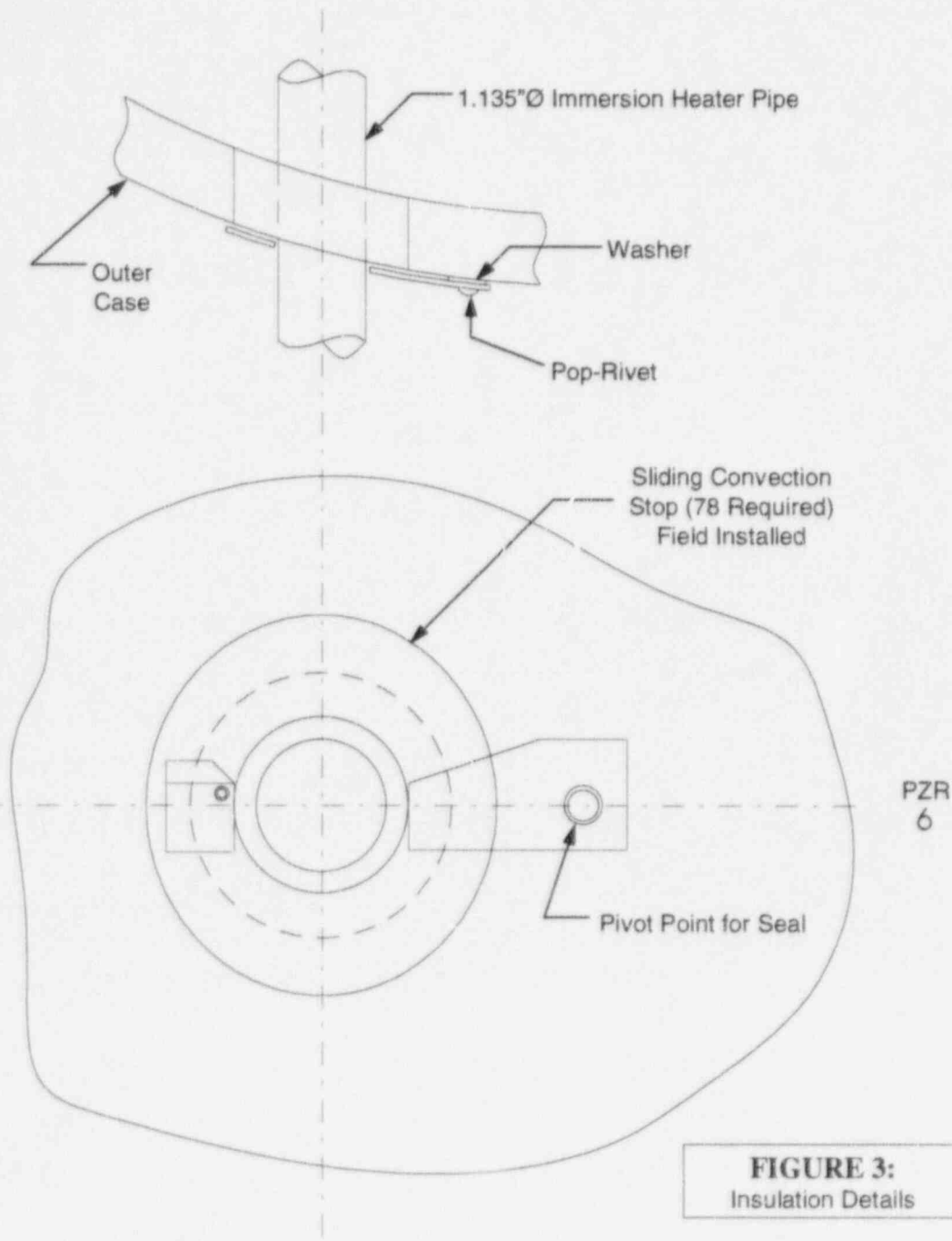


FIGURE 2:
Vessel Centerline Distance to
Surge Nozzle and
Heater Elements

- 1st Ring: 20 Heater Elements
- 2nd Ring: 26 Heater Elements
- 3rd Ring: 32 Heater Elements

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COMPONENT IDENTIFICATION

Code Classes: 2
References: Table IWC-2500-1
Examination Categories: C-B
Item Numbers: C2.32
Description: Limited Volumetric Examination Of Residual Heat Removal Heat Exchanger Nozzle-to-Shell Welds
Component Numbers: Vessels; 1RH02AA & AB, and 2RH02AA & AB
Nozzles; RHXN-01 and RHXN-02 (same for all vessels)
Drawing Numbers: 1RHX-1-ISI, Sheet 1 of 1 (Unit 1)
2RHX-1-ISI, Sheet 1 of 1 (Unit 2)

CODE REQUIREMENTS

Table IWC-2500-1, Category C-B, Item Number C2.32 requires a volumetric examination of Nozzle-to-Shell Welds (when inside of vessel is accessible) for Nozzles with Reinforcing Plate in Vessels > 1/2 inch Nominal Thickness per Figure IWC-2500-4(c).

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative would provide an acceptable level of quality and safety.

The Residual Heat Removal Heat Exchangers were fabricated with a nominal wall thickness of .875 inch and 14 inch diameter inlet and outlet nozzles which are .375 nominal wall thickness. As shown in Figure 1, the subject configuration is best characterized as a fillet welded nozzle with an internal reinforcement pad. The configuration is similar to that shown in Figure IWC-2500-4(c), except for internal location of the reinforcing pad. Because the inlet and outlet piping is welded to the subject nozzles, the inside of the RHR Heat Exchangers is not accessible as described in the "Parts Examined" text for Item No. C2.32.

Due to the geometrical constraints of this nozzle design, the ultrasonic examination of nozzle-to-vessel welds will not achieve the required 90% coverage per Code Case N-460 and Section XI, 1989 Edition. Alternatively, a "best effort" ultrasonic examination will be performed on one sample nozzle-to-vessel weld of each unit during the first period and a dye penetrant examination on each nozzle in each period for the ten-year interval. This alternative examination approach was approved for Interval 1 per NRC Safety Evaluation Report dated February 29, 1996, for Relief Request NR-18.

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The "best effort" ultrasonic examination will primarily be performed from the nozzle outside surface. Anticipated obstructions include the reinforcement fillet weld located directly above the nozzle-to-vessel weld. This fillet weld restricts inspection transducer movement and limits available examination angles. A 70° transducer will be used on the nozzle side of the weld to pass sound under the fillet weld to reach the lower $\frac{1}{3}T$ of the pressure retaining weld. Scanning from the vessel side of the weld will be accomplished by use of a 45° shear wave transducer on the shell surface to reach the examination volume. These scans are in the axial direction of the nozzle-to-vessel weld. Scans in the circumferential direction will be very limited due to the location of the fillet weld. This weld location does not allow the transducer to scan over the entire examination volume from the outside of the nozzle. It is estimated that 99.92% of the examination volume will be reached in the axial direction. With the limited circumferential scans (13.29%), the total scanning percentage is estimated at 56.61%. See Figure 2 of this report for scan coverage plots.

In addition, a VT-2 examination during system pressure testing per Category C-H is also performed on the Residual Heat Removal Heat Exchangers each inspection period to verify leaktight integrity of these welds.

Based on this information, reasonable assurance of the continued inservice structural integrity of the subject welds is achieved without performing a complete the Code examination.

Compliance with the applicable Code requirements can only be accomplished by redesigning and refabricating the Residual Heat Removal Heat Exchangers. Byron Station deems this course of action a hardship without a compensating increase in the level of quality and safety.

PROPOSED ALTERNATE PROVISIONS

The Code required volumetric examination will be completed to the maximum extent practical using ultrasonic examination techniques. This technique is described above in the "Basis for Relief" section. Additionally, a surface examination will be performed on each nozzle in each Inspection Period during the 2nd Inspection Interval.

PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the second inspection interval.

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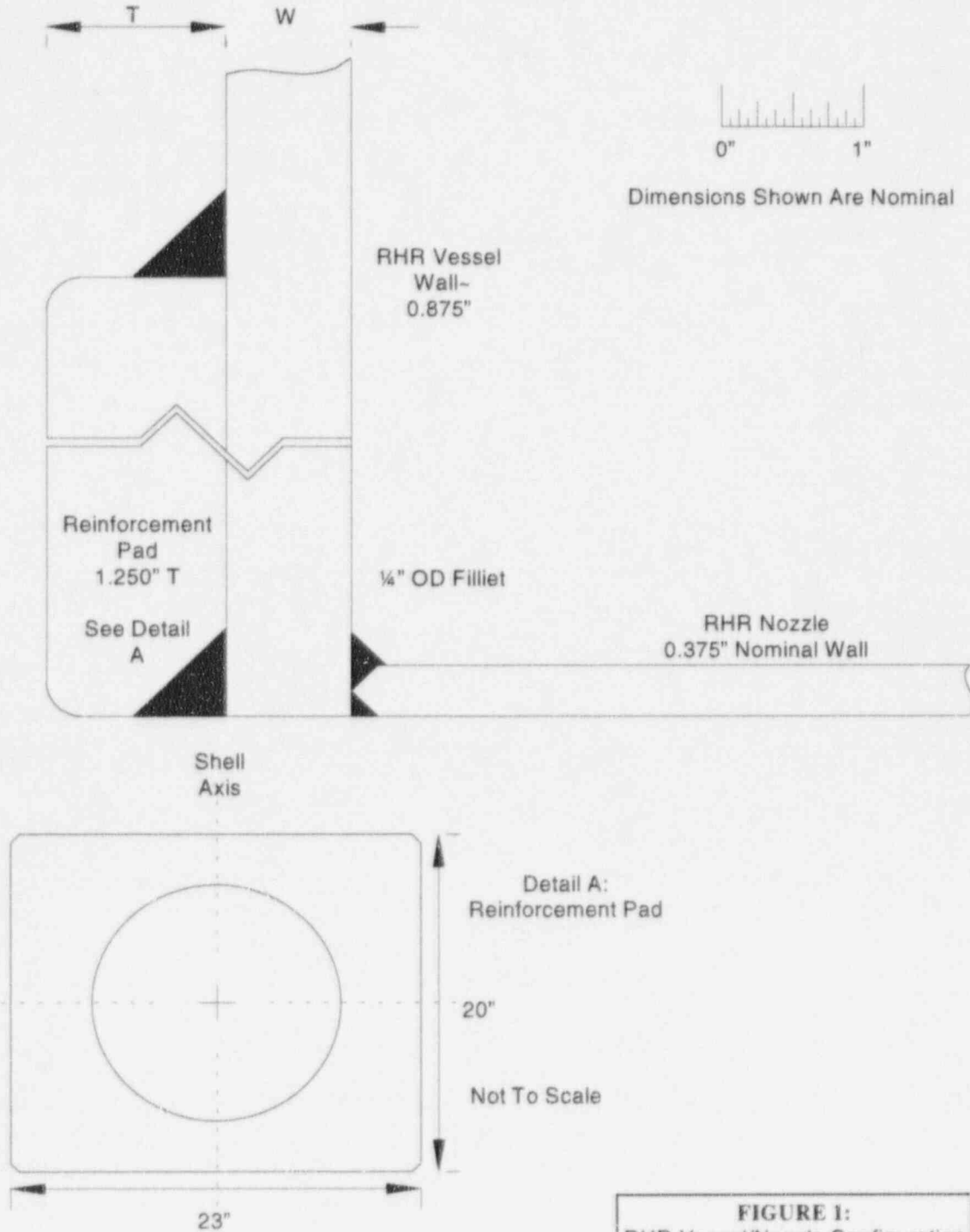
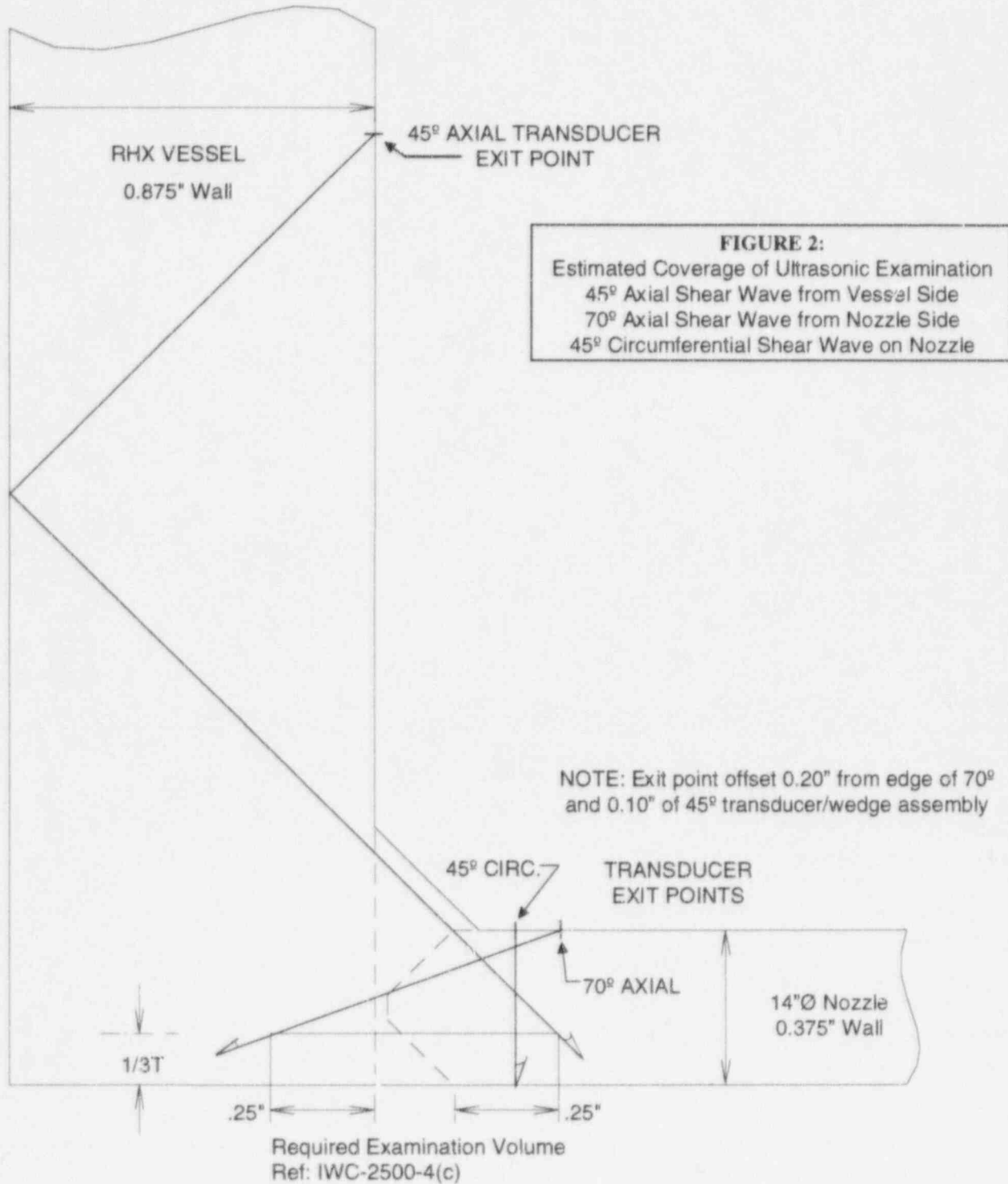


FIGURE 1:
RHR Vessel/Nozzle Configuration

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COMPONENT IDENTIFICATION

Code Classes: 1, 2 and 3
References: Table IWB-2500-1, Table IWC-2500-1, Table IWD-2500-1
Examination Categories: B-H, B-K-1, C-C, D-A, D-B, D-C
Item Numbers: B8.10, B8.20, B8.30, B8.40, B10.10, B10.20, B10.30, C3.10, C3.20,
C3.30, C3.40, D1.20, D1.30, D1.40, D1.50, D1.60, D2.20, D2.30,
D2.40, D2.50, D2.60
Description: Alternate Rules for the Selection and Examination of Class 1,
2 and 3 Integrally Welded Attachments
Component Numbers: All Class 1, 2 and 3 Integral Attachments Subject to Inservice Inspection

CODE REQUIREMENTS

Class 1 Attachments

Table IWB-2500-1, Examination Categories B-H and B-K-1 require the performance of surface or volumetric examinations, as applicable, or integral attachments with a design thickness of 5/8" or greater.

Class 2 Attachments

Table IWC-2500-1, Examination Category C-C requires the performance of a surface examination on integral attachments with a design thickness of 3/4" or greater.

Class 3 Attachments

Table IWD-2500-1, Examination Categories D-A, D-B and D-C require the performance of a visual VT-3 inspection on integral attachments.

For complete details on ASME Section XI Code examination requirements, see Tables IWB-2500-1, IWC-2500-1 and IWD-2500-1.

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BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternatives provide an acceptable level of quality and safety. Relief is being requested to allow the use of alternate requirements for the examination and selection of Class 1, 2 and 3 integral attachments detailed in Code Case N-509. The basis for this request is as follows:

- 1) During the first inservice inspection interval at the Byron Station Generating Station, no inservice flaws were detected in integrally welded attachments which would affect safety or compromise the integrity of the plant.
- 2) Within the commercial nuclear power industry, failures of integral attachments have been very rare and have not affected plant safety. When failures or inservice defects are found in integral attachments, they are usually associated with a support which has been damaged during operation. Therefore, flawed or broken integral attachments are typically detected during the investigation of damaged supports rather than during scheduled inservice inspections. One feature of Code Case N-509 is to focus the examination of integral attachments on instances where the deformation of the associated supports is identified. This requirement will increase the likelihood of locating damaged integral attachments and thereby increase the level of quality and safety provided by these alternative rules, as compared to the rules of the 1989 Edition of Section XI.
- 3) There is a significant amount of man-rem exposure and cost associated with the scheduled inspection of Class 1, 2, and 3 integral attachments.
- 4) Unlike ASME Section XI 1989 Edition, the alternate selection criteria of Code Case N-509 does not impose a minimum thickness requirement for the inspection of an integral attachment. Therefore, a greater population of integral attachments will be available for inspection because selection will not be limited to those above an arbitrary thickness. This provision improves the quality and safety level established by these examinations.
- 5) The alternate rules of Code Case N-509 provide an acceptable level of quality and safety.

PROPOSED ALTERNATE EXAMINATION

The requirements of Code Case N-509 will be used to select and examine integrally welded attachments. The selection of the integral attachments reflects 10% of the total of all nonexempt (per IWX-1220) ASME Class 1, 2, and 3 piping, pump, and valve integral attachments, and in the case of multiple vessels of similar design, function and service, one integrally welded attachment of only one of the multiple vessels. A copy of this Code Case is attached.

PERIOD FOR WHICH RELIEF IS REQUESTED

Relief is requested for the second inspection interval.



July 15, 1996

LTR: BYRON-96-0202
FILE: 3.11.0321

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Byron Station First Ten Year Interval Inservice Inspection Program
Relief Request NR-20

Byron Nuclear Power Station, Units 1 and 2
NRC Docket Numbers: 50: 454 and 455

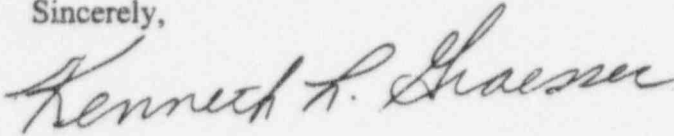
Pursuant to 10CFR50.55a(6)(ii)(A), Commonwealth Edison Company (ComEd) is providing the attached information regarding the ultrasonic examination of reactor vessel welds for Byron Station Units 1 and 2. 10CFR50.55a(g)(6)(ii)(A) requires that licensees submit information to the commission when it is determined they are unable to satisfy the requirements for augmented reactor vessel shell weld examinations. The augmented reactor vessel weld examinations are performed in accordance with ASME Section XI at the Byron Station, Units 1 and 2. 10CFR50.55a(g)(5)(iii) stipulates that when the licensee determines that examinations required by the code or addenda are impractical, information be supplied to the commission to support the determinations. Therefore, pursuant to 10CFR50.55a(g)(5)(iii), this information is being provided in the form of a relief request (NR-20).

Relief request NR-20 proposes to revise Byron Nuclear Power Station, Units 1 and 2 inservice inspection requirements for volumetric examination of the following reactor vessel welds: circumferential shell and circumferential head. Byron Station has recently completed the Unit 1 reactor vessel weld volumetric examinations during refuel outage B1R07. Due to physical and geometrical limitations, 100% ultrasonic examination of the welds was not possible. The Unit 2 reactor vessel exams are scheduled for B2R07 (Spring 1998). Since the design of the Byron Units 1 and 2 reactor vessels is identical, similar coverage percentages are expected for Unit 2.

ComEd respectfully requests that the Nuclear Regulatory Commission (USNRC) review and approve the attached relief request based on the geometric limitations and supporting justifications provided.

Please address any comments or questions regarding this request to Marcia Lesniak, Nuclear Licensing Administrator, at (708) 663-6484.

Sincerely,

A handwritten signature in cursive script that reads "Kenneth L. Graesser".

K. L. Graesser
Site Vice President
Byron Station

Attachment

cc: G. Dick, Byron Project Manager-NRR
H. Miller, Regional Administrator-RIII
H. Peterson, Senior Resident Inspector-Byron
Office of Nuclear Safety-IDNS

**BYRON STATION UNITS 1&2 FIRST INTERVAL
ISI PROGRAM PLAN-NRC SUBMITTAL
RELIEF REQUEST NR-20
(Page 1 of 7)**

COMPONENT IDENTIFICATION:

Code Class: 1
Reference: IWB-2500, Table IWB-2500-1
Examination Category: B-A
Item Numbers: B1.11 and B1.21
Description: Limited Volumetric Examination of Reactor Vessel Circumferential Shell, and Circumferential Head Welds
Component Numbers: RPVC-WR16, RPVC-WR29 (Same for both units)
Drawing Numbers: IRPV-1-ISI (Unit 1) and 2RPV-1-ISI (Unit 2)

CODE REQUIREMENT:

Table IWB-2500-1, Examination Category B-A, Item Numbers B1.11 and B1.21 require a 100% volumetric examination of the Reactor Vessel Circumferential Shell and Circumferential Head welds as detailed in Figures IWB-2500-1 and IWB-2500-3.

HISTORY:

Byron Station, during Refuel Outage B1R07, conducted ultrasonic examinations of the Byron Unit 1 Reactor Vessel. This was the last refuel outage of the third period of the first Inservice Inspection Interval and occurred in April through June 1996. Framatome Technologies Inc. (FTI) was contracted to perform the examinations with their state-of-the-art "URSULA" manipulator and their "ACCUSONEX" UT system. The examinations were performed in accordance with the requirements in ASME Section XI, Article IWA-2232, USNRC Regulatory Guide 1.150 and 10CFR50.55 a (g)(6)(ii)(A). The examination scope included 100% of the Reactor Shell welds, Head welds, and Shell to Flange welds (ASME Section XI Table IWB-2500-1, Category B-A), all 8 Reactor Nozzle to Vessel welds and Inner Radius sections (Category B-D), and Reactor Flange threads (Category B-G-1). Byron Station Reactor Vessel(s) do not have any longitudinal shell welds.

The examinations revealed 4 minor flaws, all within the acceptance criteria of ASME Section XI, Article IWB-3500.

Previously granted relief request NR-1 for Byron Station Units 1 and 2 First Inspection Interval for the Reactor Shell welds was subsequently revoked in 10CFR 50.55 a(g)(6)(ii)(A) with respect to examination coverages. During the performance of the B1R07 examinations, physical obstructions and geometry prevented UT coverage in excess of 90% of the required volume for the above listed component numbers. Full 100% UT coverage was obtained for the reactor shell welds WR-7, WR-18, and WR-34. The limited amount of examination coverage attained for shell welds WR-16 and WR-29 is provided below in the Basis for Relief.

FTI is contracted to perform the 10 year Reactor Examinations on Byron Unit 2 during the late Fall of 1997 and, as the two Byron reactors are identical, similar coverage percentages are expected.

BASIS FOR RELIEF:

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

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A. REACTOR VESSEL CIRCUMFERENTIAL SHELL WELDS

The examination of the Unit 1 Lower Shell Course-to-Dutchman weld, RPVC-WR29, is restricted by six (6) core barrel locating lugs welded to the inner surface of the vessel approximately 4 inches above the weld (See Figure 1). These lugs obstruct the automated UT inspection tool from examining the code required volume of the weld below each lug (156°). The FTI "URSULA" tool has a 6 degree movement arm and the physical size of the lugs and the "yaw" joint of the tool prevented scanning below the lugs back into the weld and surrounding base metal. All weld metal can be examined from both sides where access is available between the lugs (204°). Examinations for perpendicular and parallel reflectors covered areas accounting for 57% of the weld metal and heat affected zone (HAZ). Similarly, 57% of the weld metal can be examined for transverse reflectors from two opposing directions.

The examination of the Unit 2 Lower Shell Course-to-Dutchman weld, RPVC-WR29, is similarly obstructed (see Figure 1). All weld metal can be examined from both sides where access is available between the lugs (204°). Examinations for perpendicular and parallel reflectors can cover areas accounting for 57% of the weld metal and heat affected zone (HAZ). Similarly, 57% of the weld metal can be examined for transverse reflectors from two opposing directions.

B. REACTOR VESSEL LOWER HEAD CIRCUMFERENTIAL WELDS

The examination of the Unit 1 Lower Disk-to-Dutchman weld, RPVC-WR16, is restricted by the 58 instrument tubes that penetrate the lower disk and physically obstruct the UT search unit and/or the search unit position device as shown in Figure 2 and 3. Approximately 19% of the weld length cannot be examined. The weld and the HAZ received essentially 100% coverage for parallel reflectors from the Dutchman side and for transverse reflectors in two opposing directions. Partial coverage is achieved for parallel reflectors from the disk side on the remainder of the weld resulting in an aggregate of all scan coverage of approximately 81%.

The examination of the Unit 2 Lower Disk-to-Dutchman weld, RPVC-WR16, is similarly restricted, see Figure 2 and 3. The weld and the HAZ will receive essentially 100% coverage for parallel reflectors from the Dutchman side and for transverse reflectors in two opposing directions. Partial coverage can be achieved for parallel reflectors from the disk side on the remainder of the weld resulting in an aggregate of all scan coverage of approximately 81%.

C. REACTOR VESSEL SHELL-TO-FLANGE WELDS

The examination of the Unit 1 Nozzle Shell Course-to-Flange weld, RPVC-WR7, is located immediately below the tapered portion of the flange which previously prevented 100% examination of the adjacent base metal weld (See Figure 4) based on Preservice Inspection (PSI) results. Coverage percentages indicated were not expected to improve with current examination techniques, however, automated examinations performed in B1R07, Spring 1996, achieved 100% coverage of required examinations. All of the code required weld volume and the adjacent base metal was examined for parallel reflectors.

The examination of the Unit 2 Nozzle Shell Course-to-Flange weld, RPVC-WR7, is similarly obstructed by the vessel flange taper per Figure 4. All of the code required weld volume and the adjacent base metal can be examined for parallel reflectors as shown in Figure 4.

Strict ASME Section III quality controls were used when designing, fabricating, and installing these welds. In addition, these welds were volumetrically examined during Preservice Inspections with no irregularities found.

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For welds referenced in Sections A, B, and C above, the probability of a flaw occurring only in one of the areas not being examined is extremely small. Most future indications of significant size will be found by the examination of the weld as it is currently performed.

PROPOSED ALTERNATE EXAMINATION:

The ultrasonic examination of the Byron Unit 1 reactor vessel was performed to the maximum extent possible. No alternative volumetric examination is proposed to examine the areas not scanned due to obstructions or geometric constraints.

VT-1 inspection was conducted on the weld(s) and HAZ(s) from the inside clad surface utilizing a submersible robot during the Byron Unit 1 Refuel Outage B1R07. Additionally, a VT-2 examination during system pressure testing per Category B-P is performed on the Reactor Vessel each refueling outage to verify leaktight integrity of these welds.

The ultrasonic examination of the Byron Unit 2 reactor vessel will also be performed to the maximum extent possible. The same obstructions and geometric constraints are expected to limit the examination. The visual examinations will be repeated for the Unit 2 RPV welds.

JUSTIFICATION:

The Code required volumetric examination has been completed to the maximum extent practical using ultrasonic examination techniques for Byron Unit 1. The RPV examinations are conducted using an automated technique from the I.D. of the vessel. Access to allow inspection from the O.D. (shell side) of these welds is restricted due to the structural concrete surrounding the vessel.

Reasonable assurance of the continued inservice structural integrity of the subject welds is achieved without performing a complete Code examination. The weld(s) have received visual examinations (VT-1 and VT-2) to visually verify the integrity of the welds.

Compliance with the applicable Code requirements can only be accomplished by redesigning and refabricating the Reactor Vessel(s) and/or building a structure surrounding the vessel(s). Byron Station believes this course of action is a hardship without a compensating increase in the level of quality and safety.

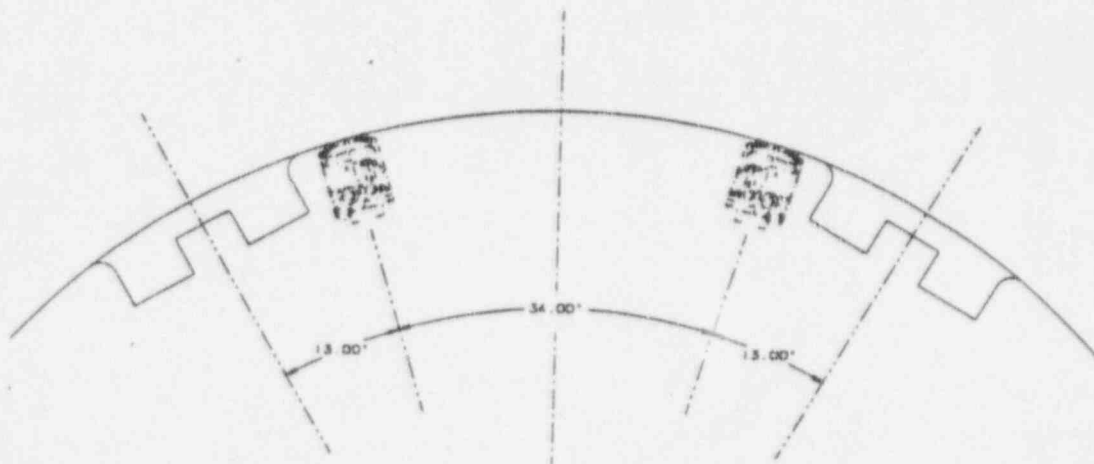
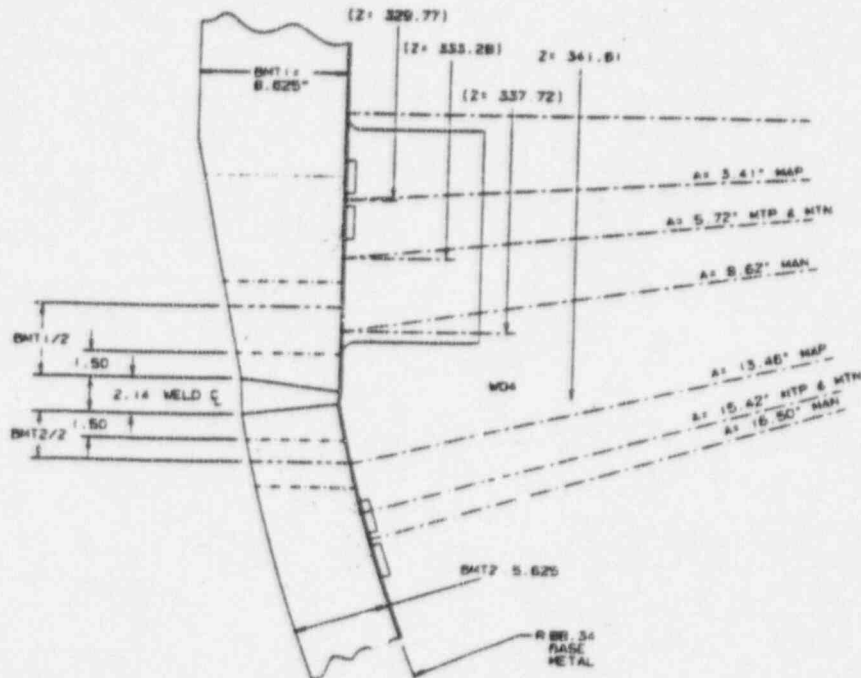
The upcoming Byron Unit 2 Reactor Vessel examination will be performed to the maximum extent practical using the same techniques and examination methods as used for Unit 1.

PERIOD FOR WHICH RELIEF IS REQUESTED:

Relief is requested for the first 10 year inspection interval for Byron Units 1 and 2.

BYRON STATION UNITS 1&2 FIRST INTERVAL
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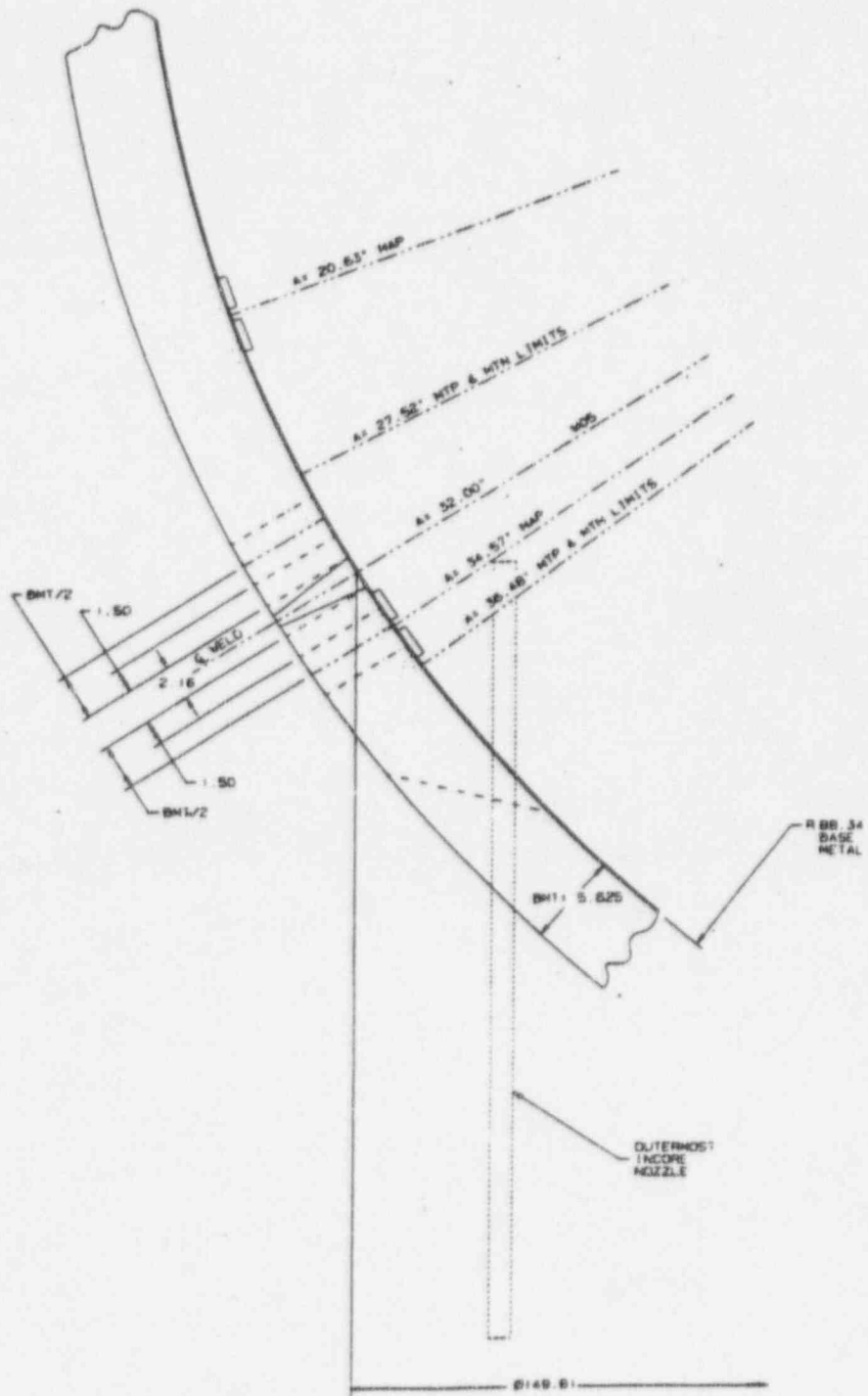
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 Figure 1



NOTE: THIS PARTIAL PLAN
 VIEW IS 1/8 SCALE

GEOMETRY IN THE E-FILE IS 1/2 FULL SIZE

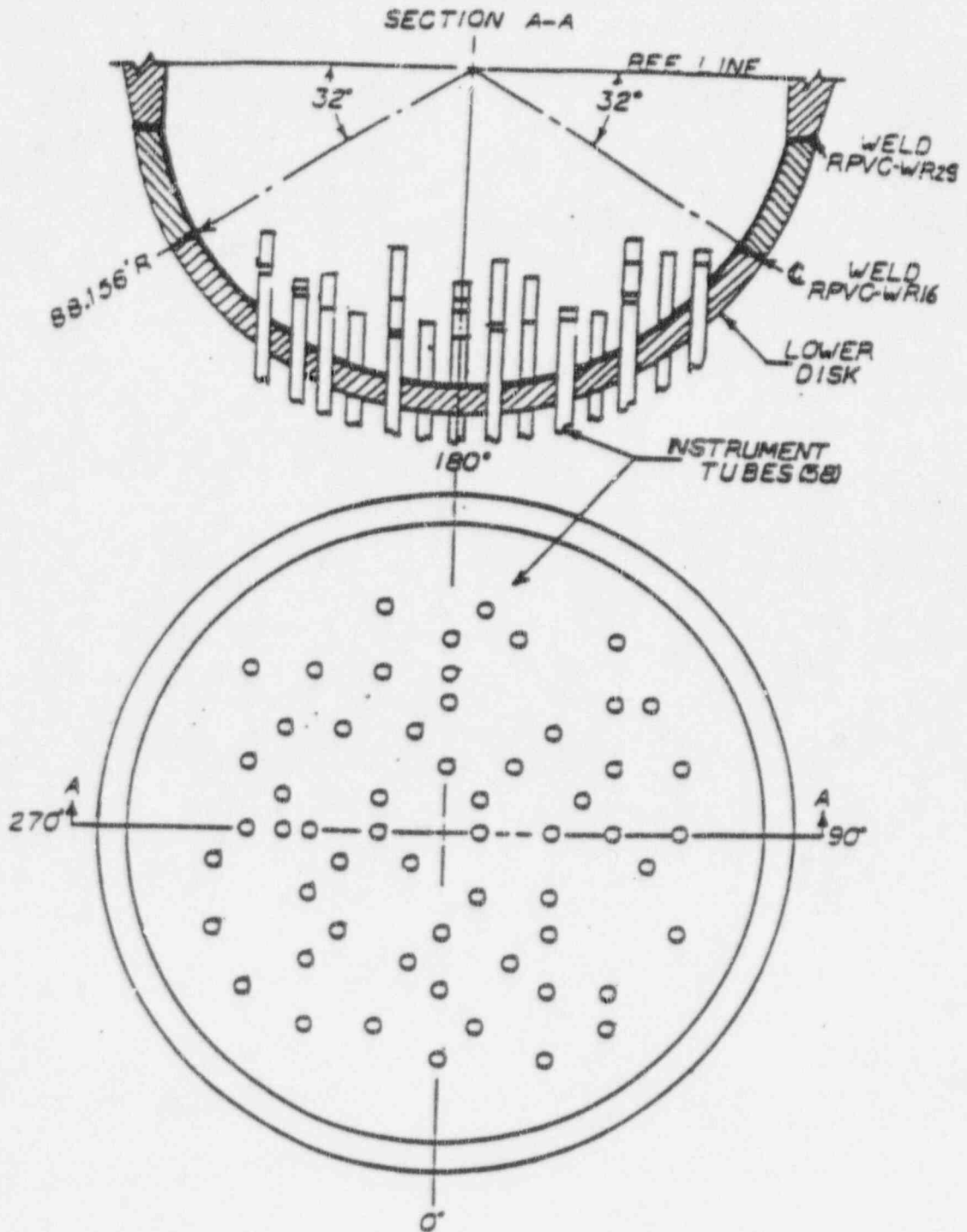
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Figure 2



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Figure 2



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 Figure 4

