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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Response to Request for Additional Information Regarding the
Third 10-Year Interval Inservice Inspection (ISI) Relief Requests

Ladies and Gentlemen:

On July 10, 2006, Southern Nuclear Operating Company (SNC) submitted relief requests for the Edwin I. Hatch Nuclear Plant – Units 1 and 2, Third 10-Year Interval ISI Program. By facsimile letter on February 5, 2007, the NRC requested additional information concerning these relief requests.

The enclosures to this letter contain the SNC response to the referenced NRC Request for Additional Information (RAI).

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

B. J. George
Manager, Nuclear Licensing

BJG/MNW/daj

Enclosures:
1. SNC Response to Request for Additional Information
2. Figures
3. Revised Relief Request RR-51
4. Synopsis of Leak Detection and Radiation Monitoring

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
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RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

Enclosure 1

Edwin I. Hatch Nuclear Plant
Response to Request for Additional Information Regarding the
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SNC Response to NRC Request for Additional Information

HATCH NUCLEAR PLANT
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
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NRC Request Pertaining to RR-42 and RR-43

For RR-42 the licensee requested relief from the volumetric examination requirements in ASME Code, Section XI, Category B-A, Item B1.21 and Item B1.22 for Hatch, Unit 1 reactor pressure vessel (RPV) bottom head meridional welds BHD-A thru F and circumferential weld C.

For RR-43 the licensee requested relief from the volumetric examination requirements in ASME Code, Section XI, Category B-A, Item B1.21 and Item B1.22 for Hatch, Unit 2 RPV bottom head dome meridional welds 2BHD-A through D and bottom head torus meridional welds 2BHT-A through G.

The ASME Code, Section XI, Table IWB-2500-1, Category B-A, Item B1.21 and B1.22 requires a volumetric examination of essentially 100% of the accessible length of all welds on the subject RPV bottom head.

According to the information provided by the licensee in its letter dated July 10, 2006, the licensee was unable to obtain any volumetric coverage on Hatch, Unit 1 RPV bottom head dome meridional welds 2BHD-A through D and circumferential weld C due to the RPV support skirt causing the subject welds to be inaccessible for examination. For Hatch, Unit 1, the licensee examined bottom head meridional welds BHT-A, BHT-B, BHT-C, BHT-D, BHT-E, BHT-F, BHT-G, and BHT-H, and bottom head circumferential weld C-7 in 2006 and the licensee obtained 100% coverage for each weld. The licensee detected no indications during these examinations.

For Hatch, Unit 2, the licensee found the examinations to be limited for bottom head dome meridional welds 2BHD-A through D and bottom head torus meridional welds 2BHT-A through G due to the control rod drives that penetrate the bottom head near the welds and the RPV support skirt interfering with the examinations. The licensee obtained 27% to 28% of the length of welds bottom head dome meridional welds 2BHD-A through D. In addition the licensee obtained 88% of the length of each weld for the bottom head torus meridional welds 2BHT-A through G. The licensee detected no indications during these examinations.

For this particular ASME Code, Section XI, Examination Category B-A and Item Numbers B1.21 and B1.22, the ASME Code specifically states examination is of the accessible weld length. Therefore, the licensee may want to consider reassessing the ASME Code requirements in that if the licensee actually obtained essentially 100% of the accessible weld length relief may not be required.

SNC Response

RR-42 and RR-43 are withdrawn because the accessible length of the welds was examined.

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NRC Request Pertaining to RR-44

The licensee requested relief from volumetric examination requirements in ASME Code, Section XI, Table IWB-2500, Category B-D, Item B3.90 for nozzle-to-vessel welds listed in the licensee's Tables RR-44-1 and RR-44-2 for Hatch , Units 1 and 2 respectively.

1. **Provide drawings or sketches of the ASME Code, Section XI, Category B-D components listed in Tables RR-44-1 and RR-44-2 for Hatch, Units 1 and 2 respectively. Drawings or sketches of every component are not required. An example of one each type of similar components will satisfy this request.**

2. **For the nozzles listed in Tables RR-44-1 and RR-44-2, identify which nozzle belongs to which vessel.**

3. **Clarify if the examinations of the nozzles listed in Tables RR-44-1 and RR-44-2 were performed from the inside or the outside of the nozzles.**

4. **Clarify which ASME Code Edition and Addenda were used regarding the Performance Demonstration Initiative (PDI) examination techniques.**

5. **Describe the plant's leakage and radiation monitor systems with regard identifying leakage from the subject welds. Also, describe when VT-2 visual examinations were last performed on the subject welds.**

SNC Response

1. Figures 1, 2, 3, 4, 5, 6, 7, and 8 are provided in Enclosure 2.

2. Table RR-44-1 is for Unit 1 (weld numbers are prefixed by 1B11) and Table RR-44-2 is for Unit 2 (weld numbers prefixed by 2B11).

3. The examinations were performed from outside the nozzles.

4. ASME Section XI, Supplement 7, *Qualification Requirements for Nozzle-to-Vessel Welds*, was implemented on November 22, 2002, as required by 10 CFR 50.55a(g)(6)(ii)(C). Examinations performed at Hatch after the implementation of Supplement 7 were performed in Spring 2005 and Spring 2006; therefore, the 2001 Edition of Section XI was used regarding the Performance Demonstration Initiative (PDI) examination techniques.

5. The leakage and radiation monitoring systems for identifying leakage from these welds, which are located in the drywell, are discussed in Enclosure 4. VT-2 visual examinations are performed as part of the leakage test conducted each refueling outage.

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NRC Request Pertaining to RR-45

The licensee requested relief from the surface examination requirements in ASME Code, Section XI, Code Case N-509, Category B-K, Item B10.10 for Hatch, Unit 2 RPV stabilizer brackets SB1 through SB6.

Provide drawings of the RPV stabilizers.

SNC Response

Figure 11 is provided in Enclosure 2.

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NRC Request Pertaining to RR-51

The licensee requested relief from volumetric examination requirements in ASME Code, Section XI, IWB-2500-1, Category B-A, Item B1.12 for RPV longitudinal welds, shown in the licensee's Table RR-51-1 for Hatch, Unit 1.

1. **Provide drawings or sketches of the ASME Code, Section XI, Category B-A, Item B1.12 components listed in Table RR-51-1 for Hatch, Unit 1.**

2. **Describe the plant's leakage and radiation monitor systems with regard identifying leakage from the subject welds. Also, describe when VT-2 visual examinations were last performed on the subject welds.**

3. **For weld C-4-B, has the licensee considered VT-1 visual examination by remote camera or any other type of visual examinations of the subject weld as an alternative to the ASME Code volumetric examinations if they can not be performed in the February 2008?**

4. **Has the licensee considered examining weld C-4-B from the inside of the RPV if at any time there is a core off load and dismantling of internal components for Hatch, Unit 1?**

5. **In RR-51 the licensee erroneously stated that the ASME Code requirements for ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.12 requires that 100% of the accessible length of each weld be examined. Note 2 of the ASME Code, Section XI Table IWB-2500-1, Examination Category B-A, Item B1.12 requires that essentially 100% of the weld length is to be examined, not the accessible length. Please correct this error in RR-51.**

SNC Response

1. Figure 1 is provided in Enclosure 2.

2. See the response to question 5 in RR-44.

3. SNC has requested the vendor to develop tooling to perform an ultrasonic examination of C-4-B. In the unlikely event that the ultrasonic examination cannot be performed, a VT-1 examination will be performed to the extent practical.

4. The obstructing tie rod would only be removed if there was an issue with the tie rod discovered during the outage in-vessel examinations. Since discovery would be during the outage, there would be insufficient time to obtain automated scanning equipment to examine C-4-B.

5. RR-51 was revised to comply with this request and is resubmitted in Enclosure 3.

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NRC Request Pertaining to RR-58, RR-59, RR-60, and RR-62

1. **Provide drawings or sketches of the RHR heat exchangers welds for Hatch, Units 1 and 2.**

2. **Identify the materials (both base material and weld material) by material specification associated with the welds in these RRs. Discuss how the materials properties and/or nozzle geometry hindered the volumetric examinations.**

3. **Describe the plant's leakage and radiation monitor systems with respect to identifying leakage from the welds. Also, describe when VT-2 visual examinations were last performed on the subject welds.**

4. **For ASME Code, Section XI, Category C-B, Item Number C2.21 where both a volumetric and surface examination are required, please provide the coverage obtained for the surface examinations.**

SNC Response

1. Figures 9 and 10 are provided in Enclosure 2.

2. Material Properties - The RHR Heat Exchanger shell and head were fabricated from SA-516 material. The inlet and outlet nozzles were fabricated from SA-541, Class II material. The weld metal is not shown on the outline drawings and associated Bill of Materials. These materials did not have an appreciable effect on the examination. Nozzle Geometry – As shown in the figures in Enclosure 2, examinations are limited by the geometry.

3. The plant's leakage and radiation monitor systems with regard to identifying leakage from the RHR heat exchanger welds is discussed in Enclosure 4. VT-2 visual examinations are performed as part of the leakage test conducted each period.

4. The surface examination coverage for Category C-B was 100%.

Enclosure 2

Edwin I. Hatch Nuclear Plant
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Figures

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The following figures were copied from the latest examination sketches for the reactor vessels and the RHR heat exchangers. Even though a weld in a figure may have a specific identifier, the figures should be considered as generic for similar components for defining limitations.

FIGURE NUMBER	DESCRIPTION
FIGURE 1	HNP-1 Reactor Pressure Vessel Welds
FIGURE 2	HNP-2 Reactor Pressure Vessel Welds
FIGURE 3	Coverage looking for circumferential flaws in a typical large diameter flange type nozzle.
FIGURE 4	Coverage looking for axial flaws in a typical large diameter flange type nozzle.
FIGURE 5	Impact of the insulation support ring on the recirculation system outlet nozzles
FIGURE 6	Coverage looking for circumferential flaws in a typical medium diameter barrel type nozzle
FIGURE 7	Coverage looking for axial flaws in a typical medium diameter barrel type nozzle
FIGURE 8	Impact of the insulation support ring on the recirculation system inlet nozzles
FIGURE 9	HNP-1 RHR Heat Exchanger
FIGURE 10	HNP-2 RHR Heat Exchanger
FIGURE 11	HNP-2 Stabilizer Brackets

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FIGURE 1
HNP-1 REACTOR PRESSURE VESSEL

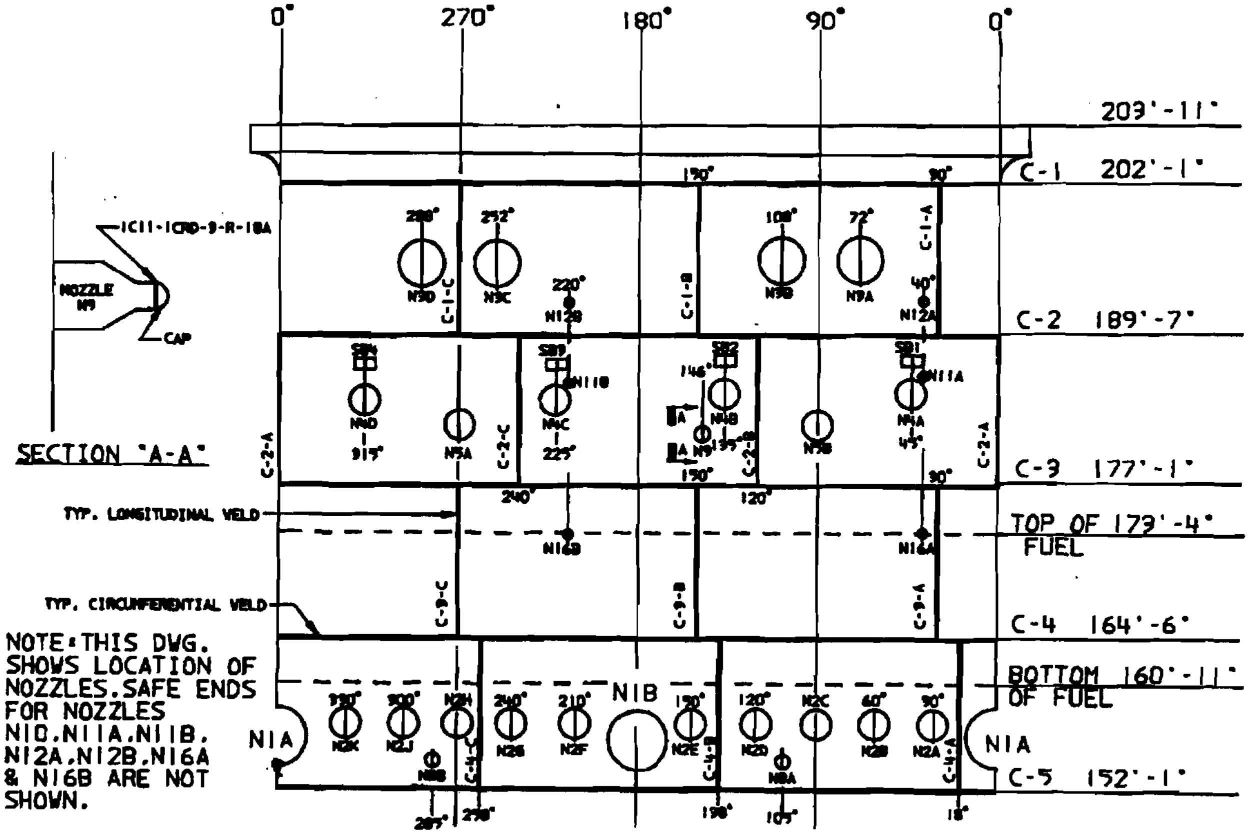

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FIGURE 2
HNP-2 REACTOR PRESSURE VESSEL

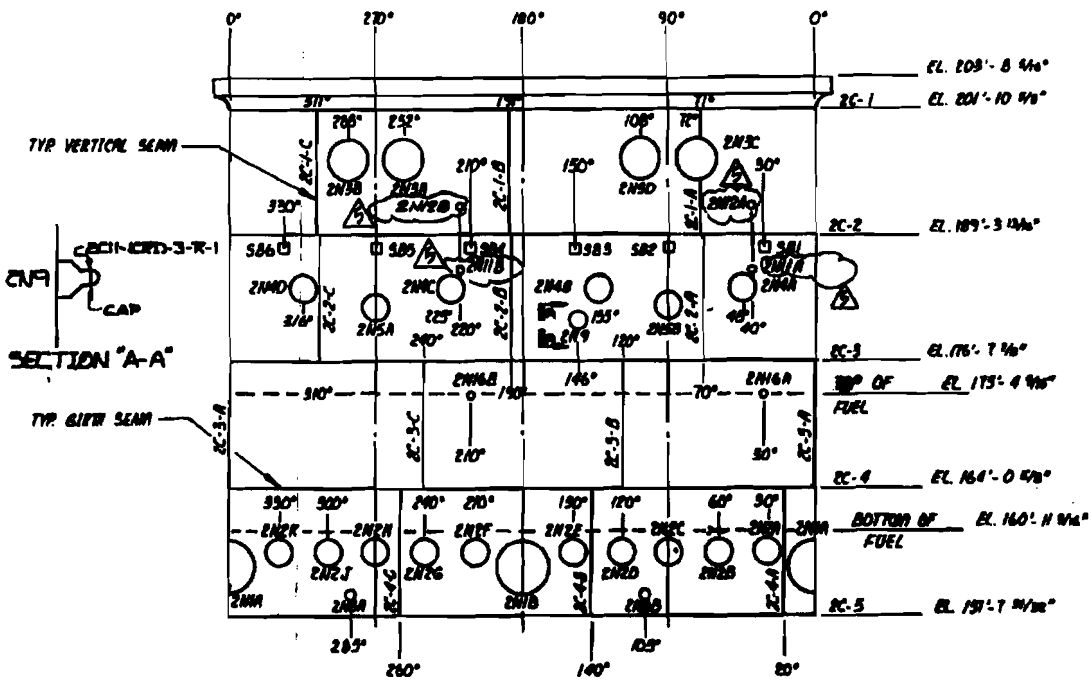

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FIGURE 3
TYPICAL LARGE DIAMETER FLANGE TYPE NOZZLE
COVERAGE LOOKING FOR CIRCUMFERENTIAL FLAWS

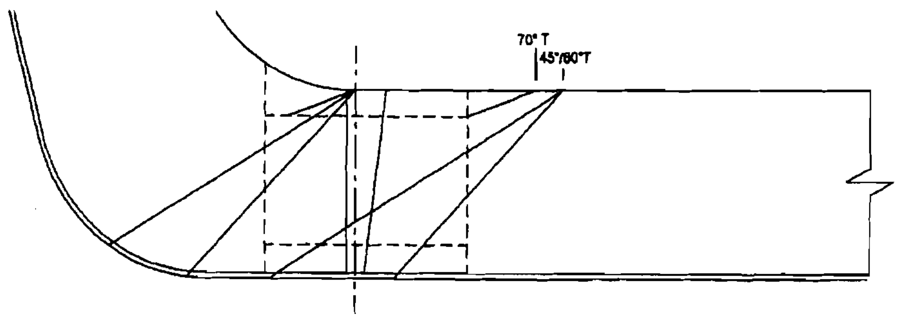

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FIGURE 4
TYPICAL LARGE DIAMETER FLANGE TYPE NOZZLE
COVERAGE LOOKING FOR AXIAL FLAWS

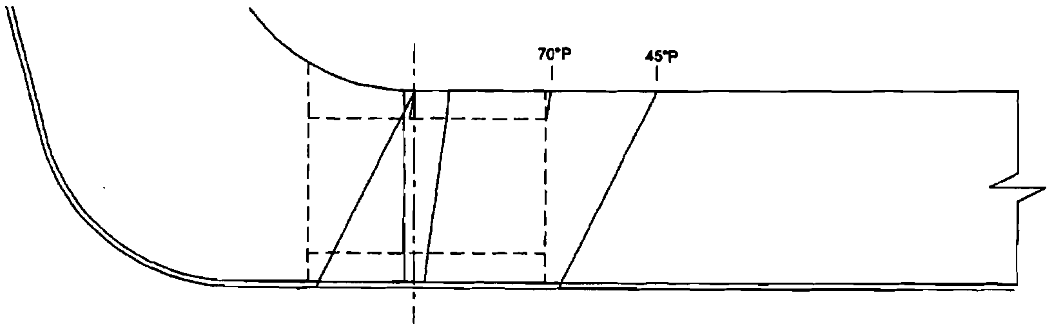

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FIGURE 5
TYPICAL COVERAGE RESTRICTION FOR RECIRCULATION OUTLET NOZZLES
DUE TO THE INSULATION SUPPORT RING

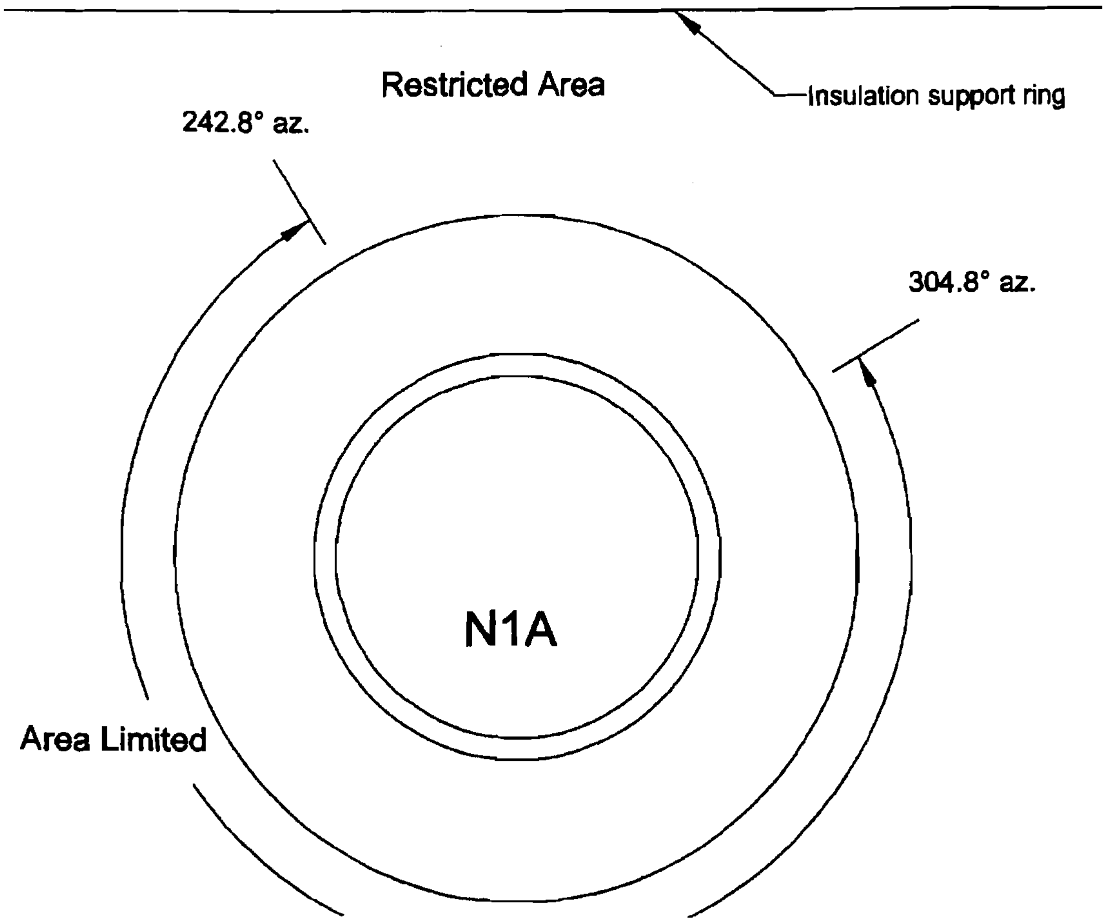

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FIGURE 6
TYPICAL MEDIUM DIAMETER BARREL TYPE NOZZLE
COVERAGE LOOKING FOR CIRCUMFERENTIAL FLAWS

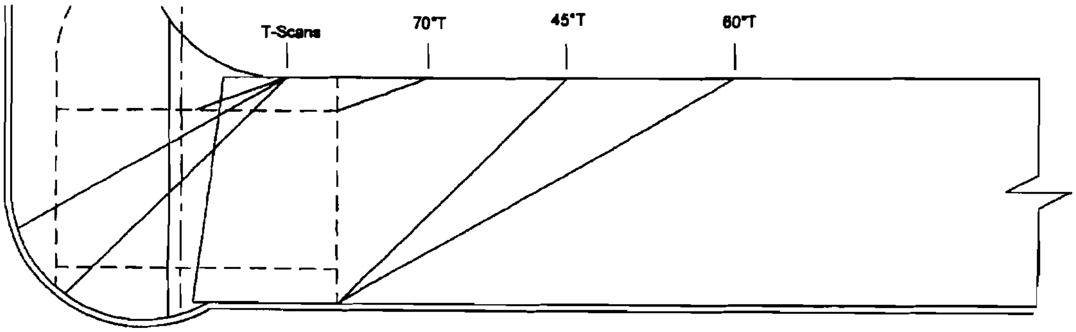

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FIGURE 7
TYPICAL MEDIUM DIAMETER BARREL TYPE NOZZLE
COVERAGE LOOKING FOR AXIAL FLAWS

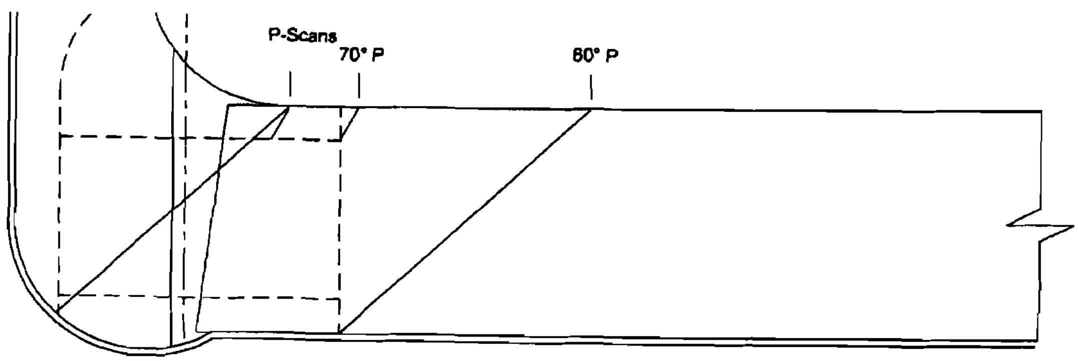

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FIGURE 8
TYPICAL COVERAGE RESTRICTION FOR RECIRCULATION INLET NOZZLES DUE TO THE INSULATION SUPPORT RING

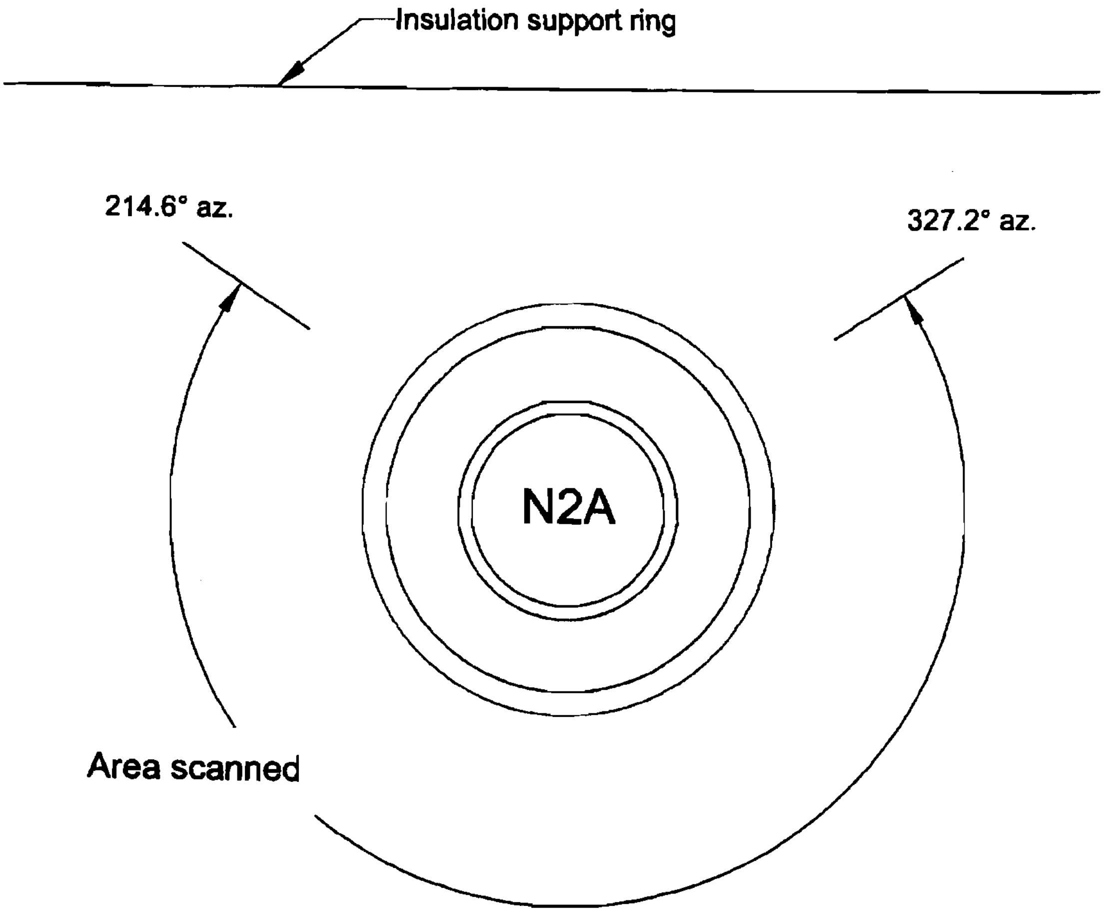

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FIGURE 9
HNP-1 RHR HEAT EXCHANGER

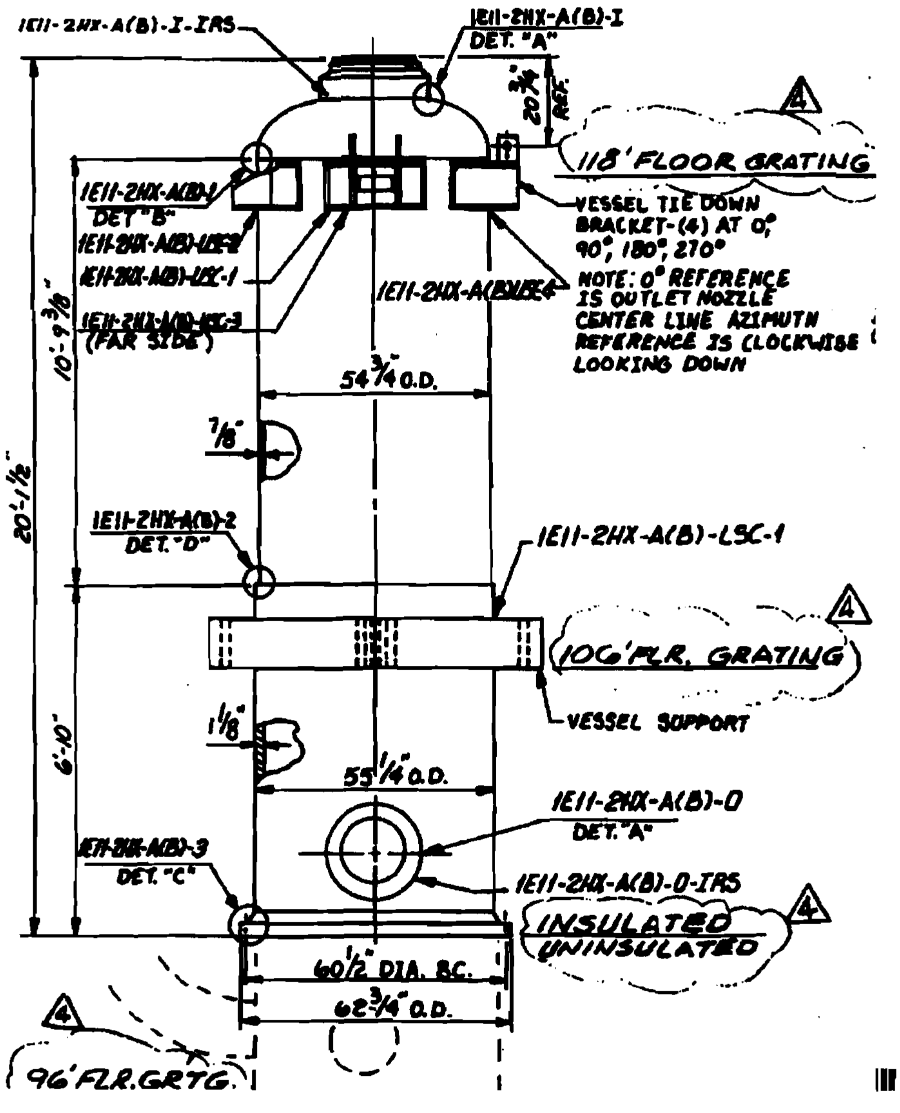

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FIGURE 10
HNP-2 RHR HEAT EXCHANGER

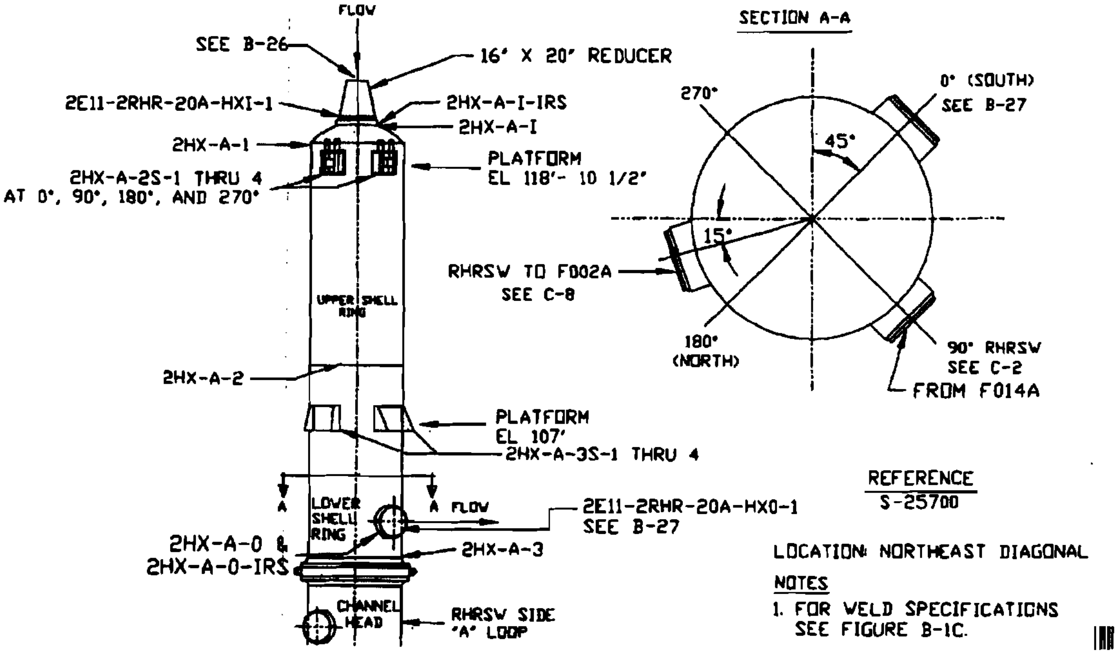

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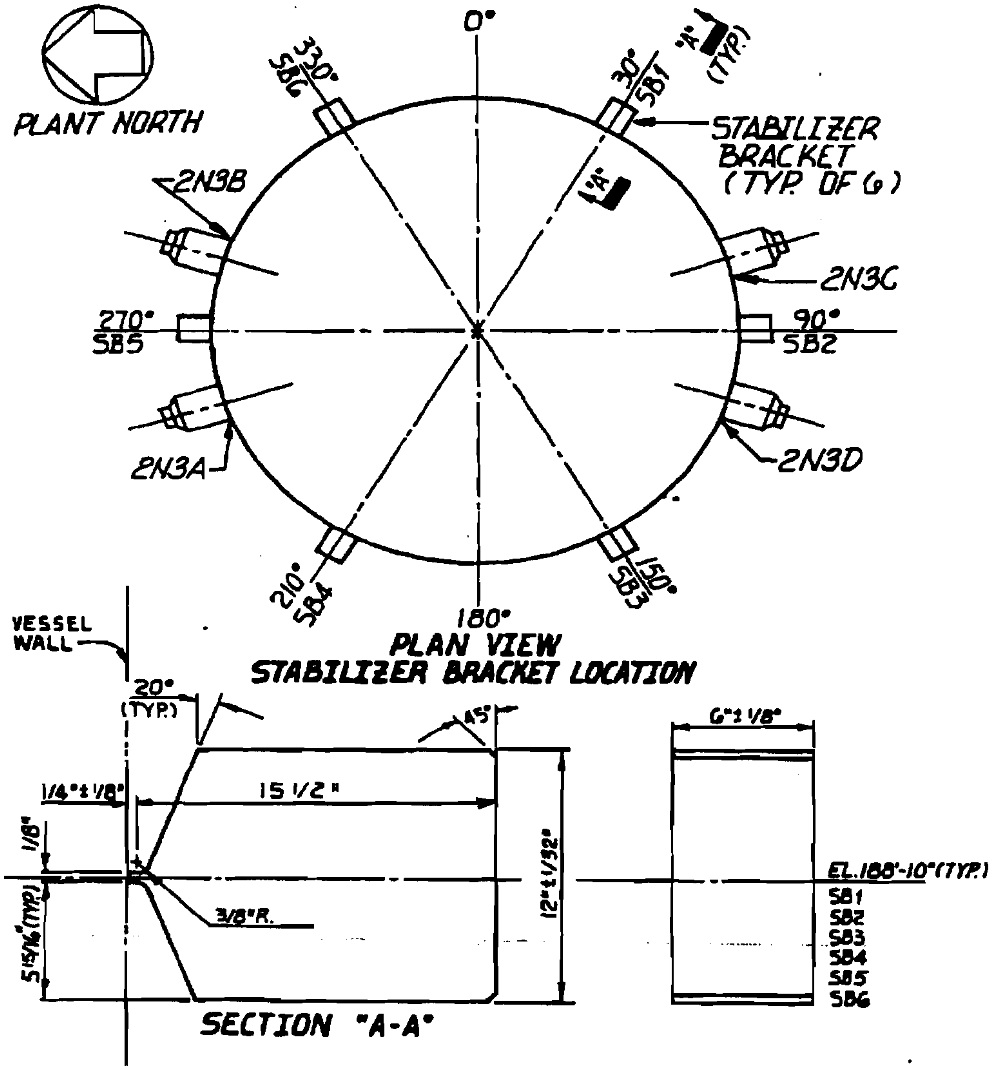
FIGURE 11
HNP-2 STABILIZER BRACKETS



Enclosure 3

Edwin I. Hatch Nuclear Plant
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Revised Relief Request RR-51

HATCH NUCLEAR PLANT
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RELIEF REQUEST RR-51

Plant Site-Unit: Edwin I. Hatch Nuclear Plant-Unit 1.

Interval-Interval Dates: 3rd ISI Interval-January 1, 1996 through December 31, 2005.

Requested Date for Approval and Basis Approval is requested by December 31, 2006 to close-out 3rd Interval activities.

ASME Code Components Affected: Class 1, ASME Section XI Category B-A, Item B1.12 reactor pressure vessel (RPV) longitudinal welds, as shown in Table RR-51-1.

Applicable Code Edition and Addenda: ASME Section XI, 1989 Edition with no addenda.

Applicable Code Requirements: Table IWB-2500-1, Examination Category B-A, Item B1.12 requires that 100% of the length of each weld be examined. Per Code Case N-460, coverage greater than 90% is acceptable.

Impracticality of Compliance: As shown in Table RR-51-1, coverage could not be obtained for seven welds. Appreciably increasing coverage was impractical due to the interferences described in Table RR-51-1.

Burden Caused by Compliance: Obtaining more coverage would require replacement of the RPV or the design of a new automated examination tool.

Proposed Alternative and Basis for Use: 10 CFR 50.55a(g)(6)(ii)(A)(*2*) required that licensees augment their reactor pressure vessel examination by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of subsection IWB of the 1989 Edition of Section XI. Per 10 CFR 50.55a(g)(6)(ii) (A)(*3*) licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 could defer the augmented reactor vessel examination to the first period of the next inspection interval. HNP-1, met this criteria; therefore, the augmented examinations were deferred until the 1st period of the 3rd interval. Additionally, as allowed, the augmented examination was used as a substitute for the reactor vessel shell weld examinations normally scheduled for the 3rd inspection interval.

Examination coverage was reported by letters dated January 19, 1999 and February 5, 1999. Pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) the NRC granted approval by letter from Herbert N. Berkow to H. L. Sumner, Jr. dated March 11, 1999 with the caveat that weld C-4-B be examined if the obstructing tie rod is removed or if technology became available for examination with the tie rod in place. The NRC concluded that the proposed alternative provided an acceptable level of quality and safety. (SNC will attempt to examine behind C-4-B during the examinations scheduled for February 2008 if equipment allows). Sufficient coverage was obtained during the examinations to assure the structural integrity of the welds. Therefore, relief should be granted per 10 CFR 50.55a(g)(6)(i).

Duration of Proposed Relief Request: The proposed relief request is applicable for the 3rd Interval.

Precedents: March 11, 1999 NRC Safety Evaluation for augmented RPV examinations.

References: None

Status: Awaiting NRC approval.

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TABLE RR-51-1		
Weld Number	**Coverage**	**Basis for Limited Coverage**
C-2-A	78%	OD examination. Proximity of insulation support ring.
C-3-A	45%	ID Examination. Proximity of a specimen bracket and jet pump riser braces.
C-3-B	79%	ID Examination. Proximity of jet pump riser braces and shroud modification hardware.
C-3-C	80%	ID Examination. Proximity of a specimen bracket and jet pump riser braces.
C-4-A	73%	ID Examination. Manipulator lower limit and proximity of shroud gusset plates.
C-4-B	0%	ID Examination. Proximity of shroud modification hardware (tie rod).
C-4-C	73%	ID Examination. Manipulator lower limit and proximity of shroud gusset plates.

Enclosure 4

Edwin I. Hatch Nuclear Plant
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Synopsis of Leak Detection and Radiation Monitoring

Leakage Detection System (LDS)

The LDS is described in more detail in HNP-1 FSAR Section 4.10 and HNP-2 FSAR Section 5.2.7.

Inside drywell

Drywell floor drain sump measurement monitors the normal design leakage collected in the floor drain sump. The drywell equipment drain sump measurement monitors identified leakage collected in the equipment drain sump, and is a closed system which receives leakage only from identified sources. All leakage inside the drywell will flow to one of these two sumps. The "unidentified leakage" is the portion of the total leakage rate received in the drywell sumps that cannot be attributed to pumps, valve seals, and the RPV head seal. The TS limit for unidentified leakage is 5 gpm. This value is based on, but conservatively much less than, the calculated flow (150 gpm) from a critical crack inside the drywell. The LDS is required to detect unidentified leakage of 5 gpm within one hour, but is capable of measuring much lower leakage rates. The post-accident radiation monitoring system (RMS) is part of the redundant LDS. The drywell fission products monitoring system provides a continuous air sampling of the drywell atmosphere through monitoring gross particulates, iodine, and noble gases. This system supplements the other methods and provides improved sensitivity to aid in determining the size and general source of leaks, particularly steam leaks.

Reactor Building

Outside the primary containment, each system is monitored in compartments, or rooms, so that leakage may be detected by leak detection sumps and area temperature indications. An increase in the normal rate of leakage into the floor drain sumps results in actuation of an alarm in the Main Control Room (MCR). Thermocouples in the Reactor Building rooms monitor ambient air temperature as well as temperature differential in the inlet/outlet of the normal ventilation and the standby coolers. High ambient air temperature or high differential temperature causes an alarm to annunciate in the MCR.