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Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards addl info re inservice insp program re ASME Code Section XI,in response to NRC 801229 request.Info provided re relief requests from weld exams & estimated man-rem exposures from volumetric exams.

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NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

January 20, 1981

Director of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No 50-263 License No DPR-22

Supplemental Information Concerning Inservice Inspection Program

The NRC Staff is currently reviewing our ASME Code inservice inspection program. This program is described in a report entitled, "ASME Code Section XI Inservice Inspection and Testing Program and Information Required for NRC Review of Requests for Relief from ASME Code Section XI Requirements." This program was submitted for review on March 15, 1978. Periodic revisions have been made with the last revision submitted for NRC Review on March 5, 1980.

On December 29, 1980 our Project Manager in the Division of Licensing telecopied a request for additional information related to the Monticello Section XI inservice inspection program. The purpose of this letter is to provide the requested information.

The required information is provided as an attachment to this letter. Please contact us if you have any questions related to our responses.

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

cc J G Keppler
G Charnoff
NRC Resident Inspector

Attachment

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121.1 The Monticello ISI Program contains several relief requests due to inaccessibility caused by the reactor vessel biological shield and thermal insulation.

Question: Describe in detail, and provide drawings where necessary, design and geometry of these components which make the code required examinations impractical. Discuss the measures necessary to make the welds accessible.

Answer: The Monticello RPV was constructed with a 2'3½" thick biological shield wall surrounding it, with the exception of the top eight or so feet. Between this wall and the reactor vessel shell is a space of approximately 1 foot that houses the thermal insulation. The only access areas to the reactor vessel is: 1. at the top eight feet above the biological shield wall, 2. through openings in the wall at each nozzle location and two inspection ports below the skirt weld, 3. by the control rod drives under the reactor head, and 4. from the vessel inside diameter.

The area above the biological shield wall and at the nozzle openings is farther obstructed by non-removable insulation. A good portion of the vessel insulation was not designed to be removed and therefore it was installed prior to the installation of the piping, electrical conduits, vessel stabilizers, duct work, etc.

A very thorough review was performed, using drawings, sketches, and previous examination reports, to try and locate weld areas that possibly could be inspected. It was concluded that some of the vessel welds appear to be close enough to nozzle openings for performing the examinations provided the insulation can be removed. An attempt will be made to remove or modify the insulation and to examine those areas identified on attachment 1 during the 1981 refueling outage. The dotted weld areas on attachment 1 by nozzles N1A, N8A, and N2G will be attempted to be examined during the next ten year interval. In addition, during the upcoming outage we are hopeful that radiation levels will permit access to the bottom of the vessel (between the control rod drives, skirt, and vessel). This would allow accessibility to a larger portion of the skirt to vessel weld and to the bottom head meridional and dollar plate welds.

Each of the welds that can be examined will be sketched to show the examination amount, extent and location. In addition, each area will be scheduled for examination during the next ten year interval.

Question: For the standby liquid control nozzle to safe end weld, estimate the percentage of weld you can volumetrically examine, and the additional percentage which will be examined with surface techniques.

Answer: The Standby Liquid Control nozzle and safe end protrudes through a hole in the vessel thermal insulation with a minimum of clearance. It appears, from an early sketch, that the bottom of the weld is next to and just outside the insulation. The insulation around this nozzle was not designed with removable panels. With the present design the nozzle to safe end weld is 100% accessible for ultrasonic examination from one side, and approximately 50-75% accessible for a liquid penetrant examination.

The volumetric and surface examinations required on this weld will be performed to the extent possible during the 1981 refueling outage. During this outage an attempt will be made to modify the insulation to allow 100% coverage of the weld for both examination methods. In addition, a sketch will be made that will identify any areas that remain inaccessible.

Question: Your program states that longitudinal and circumferential welds in the reactor vessel shell have internal access for volumetric examinations. Estimate the personnel exposure in man-rem to conduct these examinations.

Answer: The design of the Monticello reactor vessel internals does not allow internal access to the vessel welds either in or below the vessel region, but does give internal access to the top 20 ft. of the reactor vessel. This access allows one circumferential and two longitudinal welds to be examined that were thought to be inaccessible from the outside of the vessel. After the review was performed for the first part of this question, it was concluded that welds VCBB-4, VLCB-1 and VLCB-2 may be accessible from some nozzle opening in the biological shield wall. If these examinations can be performed from the vessel O.D., it would result in much lower exposure to personnel than an I.D. examination. Therefore, an attempt will be made during the 1981 refueling outage to examine all these areas.

Question: Provide details concerning the reactor vessel material surveillance program and discuss your compliance with Regulatory Guide 1.99 "Effects of Residual Elements on Predicted Radiation Damage to RV Materials."

Answer: On October 4, 1979 NSP submitted for NRC Staff Review General Electric Report NEDO-24197, "Monticello Nuclear Generating Plant Information on Reactor Vessel Material Surveillance Program," June, 1979.

As noted in this report, the copper content of the beltline welds is unknown. The results of the surveillance program will not provide a firm basis for use of other than the worst case shift from Regulatory Guide 1.99.

In NSP's License Amendment Request dated May 15, 1980 the worst case shift curve from Regulatory Guide 1.99 was proposed for use at Monticello.

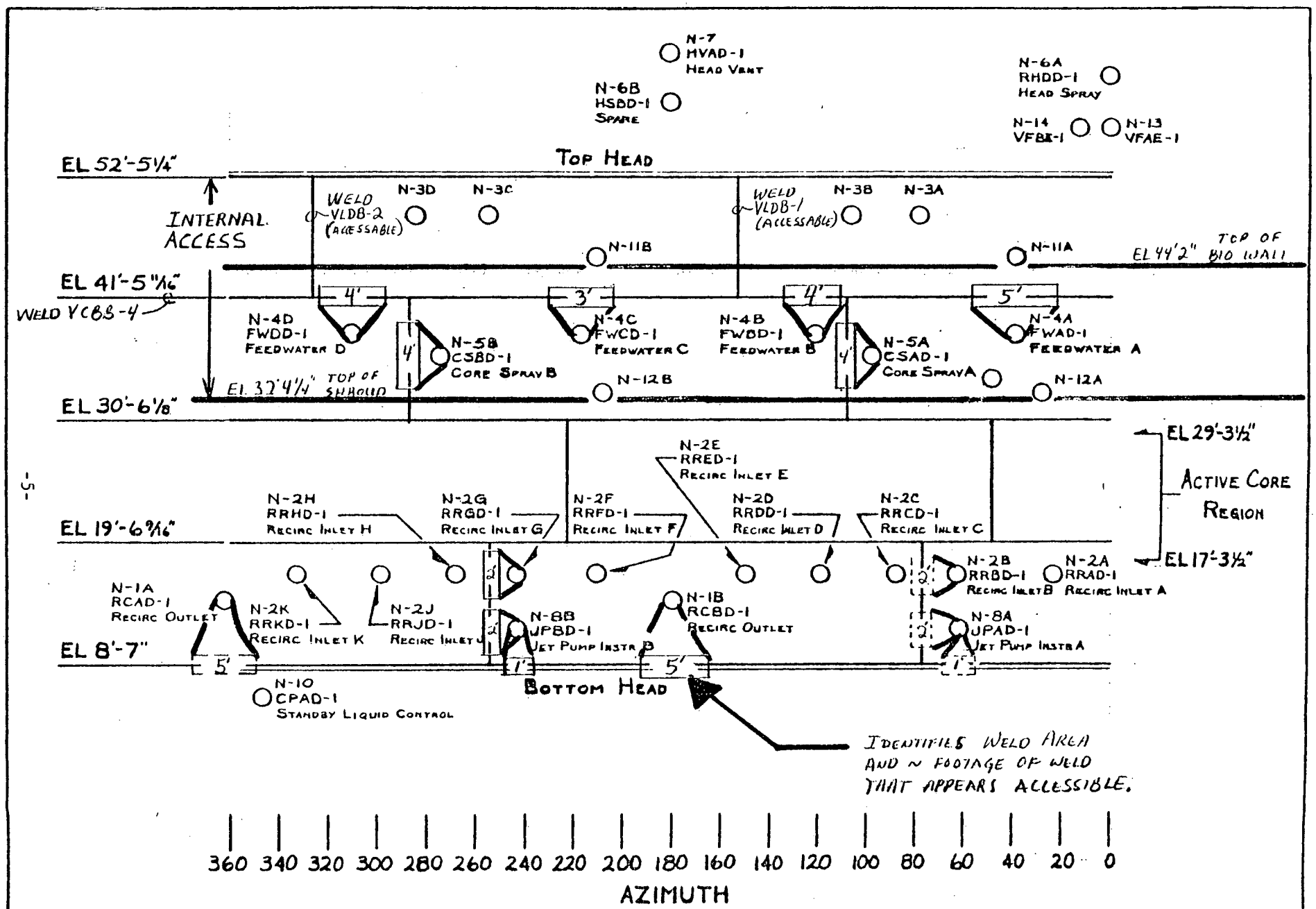
A copy of NEDO-24197 (without Appendix A and Appendix B) is attached.

121.2 Relief was requested from the Class 2 system pressure test distribution as required by the ASME Code and the requirements in the Winter 1976 Addenda were proposed as an alternative examination procedure. This Addenda is not approved by the NRC; however, the pressure test requirements in the 1977 Edition through Summer 1978 Addenda, particularly those in the Winter 1977 Addenda, may be utilized. Indicate if these alternative approved requirements will be incorporated into your program.

Response

This Request for Relief is withdrawn. The requirements of the Summer 1975 Addenda will be satisfied and 100% will be completed by the end of the next refueling outage. This outage is the last outage in the first ten year inspection interval.

The inspection program to be followed for subsequent outages will conform where possible to the 1977 Edition through the Summer 1978 Addenda of the ASME Code.



MONTICELLO UNIT 1

REACTOR VESSEL NOZZLE LOCATIONS & IDENTIFICATION

DWN: R. THORNE
CHK'D: GBS

FIGURE 4

NEDO-24197
79NED297R
Class I
June 1979
Revision 1
October 1979

MONTICELLO
NUCLEAR GENERATING PLANT

INFORMATION ON
REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

Prepared for
Northern States Power Company

by
General Electric Co.
San Jose, California

NUCLEAR POWER SYSTEMS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

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The only undertakings of the General Electric Company respecting information in this document are contained in the contract between Northern States Power Company and General Electric Company for nuclear fuel and related services for the nuclear system for Monticello Nuclear Generating Plant dated December 4, 1967 and nothing contained in this document shall be construed as changing said contract. The use of this information except as defined by said contract, or for any purpose other than that for which it is intended is not authorized; and with respect to any such unauthorized use, neither General Electric Company nor any of the contributors to this document makes any representation or warranty (express or implied) as to the completeness, accuracy or usefulness of the information contained in this document or that such use of such information may not infringe privately owned rights; nor do they assume any responsibility for liability or damage of any kind which may result from such use of such information.

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1. FLUENCE AT INNER WALL OF THE REACTOR PRESSURE VESSEL (RPV)

(By Northern States Power Company)

2. EFFECTIVE FULL POWER YEARS AS OF MARCH 31, 1977

(By Northern States Power Company)

3. RPV SUPPLIER AND SPECIFICATIONS

The Monticello reactor pressure vessel (RPV) was purchased from Chicago Bridge and Iron Company, Birmingham, Alabama (Purchase Order No. 205-55582-I, Contract No. 9-5624). The RPV is designed and constructed in accordance with the following codes and specifications:

a. ASME Boiler and Pressure Vessel Code - Section III, 1965 Edition, with addenda to and including Summer 1966 addenda

b. General Electric APED specification No. 21A1112, Rev 6

4. VESSEL BELTLINE MATERIAL

4.1 PLATE AND WELD LOCATION

Materials in the reactor pressure vessel beltline regions (as defined in 10CFR50, Appendix G, Section II.H.) are identified below as shown in Figure 1.

4.2 PLATES/FORGING AND WELDS

4.2.1 All shell plate material is ASME A-533, GR. B, Class 1, CB&I specification MS-1 dated 8/25/66, and provided by Lukens Steel Company, Coatesville, Pennsylvania.

4.2.2 Weld Materials and Process

For the Monticello vessel, the specific weld materials used in the beltline region cannot be identified. Therefore, it must be assumed that any of the weld heats provided in Appendix A could have been employed. All weld metal used on the vessel is listed.

NOTES:

1. BELTLINE AREA OF INTEREST INCLUDES:

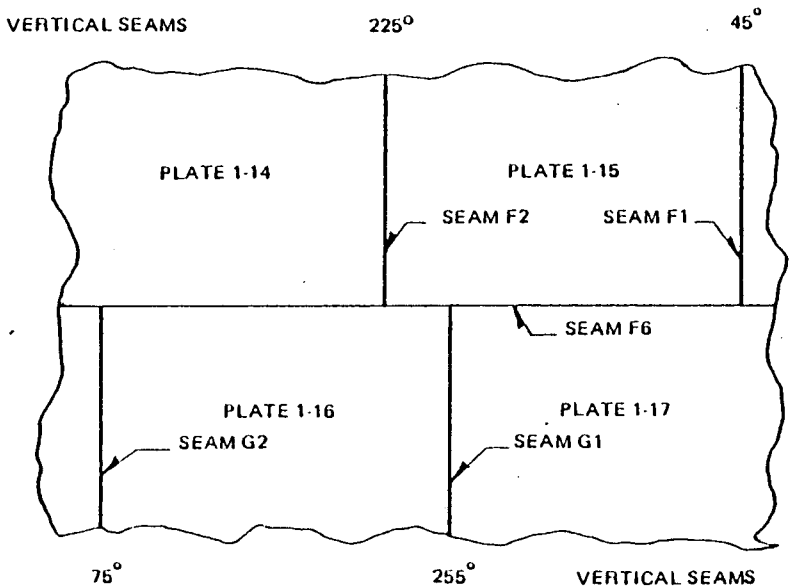
   FOUR PLATES - 1-14; 1-15; 1-16; & 1-17

   FOUR VERTICAL SEAM WELDS -
   AT 45° AND 225° FOR PLATES 1-14 & 1-15
   AT 225° AND 75° FOR PLATES 1-16 & 1-17

   ONE GIRTH SEAM AT ELEVATION 243 9/16-INCHES

REFERENCES:

1. CB&I DRAWING NO.'s 3B, 27, 28, 29 & 30 OF PO=205-55582, CONTRACT 9-5624.
2. GE REACTOR ASSEMBLY DRAWING NO. 104P919.



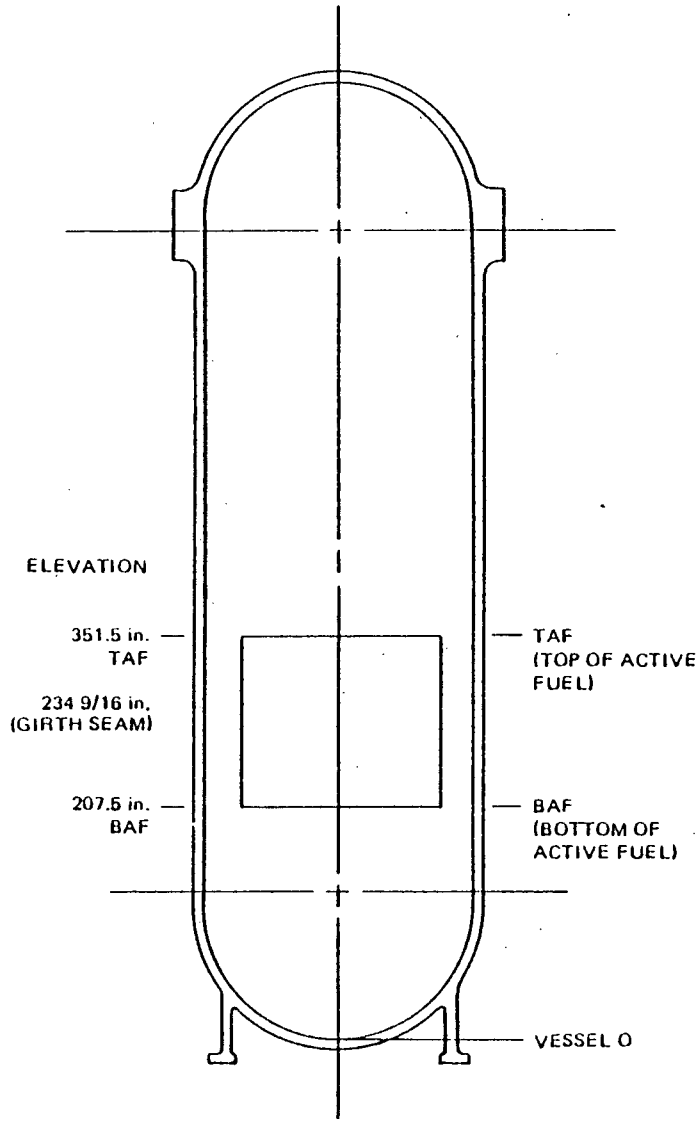


Figure 1. Monticello RPV Beltline Region

4.2.3 Lower Shell Course (Ring #1)

a. Plates (2) - 1-16 and 1-17

Piece No.	Heat No.	Slab No.
1-16	A0946	1
1-17	C2193	1

b. Vertical Weld Seams (G1 and G2)
Welding Procedure Spec - WPS-1
Coated Electrode - Type E8018NM

4.2.4 Lower Intermediate Shell Course (Ring #2)

a. Plates (2) - 1-14 and 1-15

Piece No.	Heat No.	Slab No.
1-14	C2220	1
1-15	C2220	2

b. Vertical Weld Seams (F1 and F2)
Welding Procedure Spec - WPS-1
Coated Electrode - Type E8018NM

4.3 HEAT TREATMENT

4.3.1 Plates and Test Specimens

All plates were heated 1625-1675°F, held one hour per inch minimum, and water quenched to under 400°F by holding in water for at least 3-1/2 minutes per inch of thickness, then tempered at 1200-1250°F, held one hour per inch minimum, and air cooled.

Tests from heated plates were stress relieved by heating within a rate of 75°F per hour to 1125-1175°F, held 50 hours and furnace cooled within a rate of 94°F per hour to 600°F, and air cooled.

A second temper after gas cutting was not higher than 1100°F and was held not longer than one hour per inch of thickness.

CB&I reported the following heat treatment for the plates:

Quenched and Tempered

	Mkl-15	Mkl-14	Mkl-17	Mk-16
1650°F	5 hr 20 min	5 hr 20 min		
1230°F	5 hr 20 min			
1225°F		5 hr 20 min		
1600°F			5 hr 20 min	5 hr 20 min
1220°F			5 hr 20 min	
1200°F				5 hr 20 min
Stress Relief				
1100°F	3 hr	3 hr		

Tests stress relieved at 1150°F for 50 hours.

4.3.2 Post Weld Heat Treatment

a. Lower shell course - vertical weld seams G1 and G2 were held for a total of 12 hours and 13 minutes at 1150°F.

b. Lower Intermediate shell course - vertical weld seams F1 and F2 were held for a total of 6 hours and 38 minutes at 1150°F.

4.4 CHEMICAL ANALYSES

At the time of the RPV construction, the only requirements for chemical composition were those contained in the ASTM material specifications and did not include copper (Cu). By tracing the heat numbers for the beltline plates back to Lukens Steel (the supplier), the copper contents of the plate material were obtained. The ladle analysis for % Cu was measured and retained by the supplier even though it was not required by the purchase specification.

The chemical analyses of the weld materials do not report copper, phosphorus, or sulfur content and attempts to obtain the missing weld metal information from CB&I have been unsuccessful.

The chemical analyses for vessel beltline materials are as follows:

	Weight Percent							
Description	C	Mn	P	S	Cu	Si	Ni	Mo
a. Plate								
Lower Shell Course								
1-16	0.21	1.41	0.010	0.016	0.14	0.15	0.56	0.47
1-17	0.21	1.15	0.010	0.010	0.17	0.24	0.50	0.48
Lower Int. Shell Course								
1-14	0.20	1.31	0.010	0.014	0.17	0.22	0.58	0.45
1-15	0.20	1.31	0.010	0.014	0.17	0.22	0.58	0.45

b. Weld

		Weight Percent				
Heat	Lot	C	Mn	Ni	Si	Mo
588380	E701A27A	0.04	0.96	0.97	0.35	0.50
649T217	D724A27A	0.04	0.97	0.94	0.36	0.51
78B723	D818A27A	0.06	0.87	0.90	0.26	0.55
08T401	F821A27A	0.04	0.96	0.93	0.36	0.52
659T568	H721A27A	0.06	0.93	0.86	0.36	0.50
422W1601	F813A27A	0.05	1.10	0.97	0.38	0.56
89B825	A718A27A	0.05	1.13	0.98	0.39	0.57
84B719	A703A27A	0.06	1.14	0.92	0.41	0.50
649S416	A716A27A	0.06	1.10	0.97	0.35	0.50
574972	A709A27A	0.05	0.90	0.94	0.23	0.52
977129	B822A27A	0.05	0.92	0.95	0.30	0.50
977080	A814A27A	0.05	0.92	0.95	0.30	0.50
80D225	E824A27A	0.05	0.87	0.85	0.37	0.51
659T174	C729A27A	0.06	0.95	0.99	0.34	0.52
78B743	C727A27A	0.04	1.04	0.93	0.41	0.52
78B743	C723A27A	0.04	0.98	0.92	0.41	0.45

Heat	Lot	Weight Percent C	Mn	Ni	Si	Mo
654S464	C721A27A	0.05	1.05	0.89	0.43	0.47
587784	F702A27A	0.05	0.85	0.95	0.31	0.49
659T568	H721A27A	0.05	0.87	0.91	0.29	0.48
654T486	H715A27A	0.06	0.96	0.85	0.22	0.48

4.5 TENSILE PROPERTIES (UNIRRADIATED)

Plate	Heat No.	Slab	Yield Strength (ksi)	Ultimate Strength (ksi)	Elongation (% in 2 inches)
Lower Shell Course	A0946	1	73.4/66.0	91.5/91.5 90.5/90.5	27/27
	C2193	1	66.0/63.5	89.5/87.5 87.1/87.3	28/30
Lower Int Shell Course	C2220	1	64.8/63.1	90.0/88.4 89.3/89.8	30/26
	C2220	2	65.2/62.2	91.4/90.4 87.7/88.0	31/30

4.6 UNIRRADIATED FRACTURE TOUGHNESS PROPERTIES

The ASME code and specifications applicable to the Monticello RPV did not require upper shelf measurements; therefore, no data are available. Comparison of the Monticello CVN data with impact data in the literature and from other BWR vessels indicates that the upper shelf for Monticello plates should meet the 75 ft-lb limit presently required by 10 CFR 50 Appendix G.

		Impact Properties			
a.	Plates	Drop Wt.	Longit CVN ft-lbs	Test Temp.	*RT_{NDT}
	C2193-1	0°F	37, 53, 37	+10°F	+6°F
	A0946-1	0°F	53, 42, 47	+10°F	0°F
	C2220-1	0°F	60, 93, 81	+10°F	0°F
	C2220-2	0°F	81, 33, 61	+10°F	+14°F

b. Weld Filler Material

No data are available on the fracture toughness of the weld material used in the Monticello RPV. The only information useful for estimating fracture toughness is the requirement in the purchase specification which specifies that ASME code requirements must be met. The ASME code applicable to Monticello required an average charpy V notch value of 30 ft-lb with a minimum value of 25 ft-lb for one specimen. GE purchase spec 21A1112, Rev 6, required impact tests to meet the ASME values at a temperature no higher than 40°F. In addition, the specification required an NDT temperature no higher than 40°F.

If there were no deviations, the initial reference temperature for weld filler materials determined by using methods in Design Procedure Y1006A006 is:

$$RT_{NDT} = 40^{\circ}F + (2 \times 25^{\circ}) - 60^{\circ} = 30^{\circ}F \text{ (CVN)}$$

Since 40°F is the specification value, the worst case initial RT_{NDT} would be the higher of the two values or 40°F.

A Quality Assurance verification that no deviations from the specifications occurred is in process.

*Initial reference temperatures (RT_{NDT}) were derived from available data in accordance with the methods of Design Procedure Y1006A006 (see Appendix B).

5. VESSEL MATERIAL SURVEILLANCE PROGRAM

Very little data are available on surveillance specimens. The information supplied states that surveillance samples were fabricated from beltline plate 1-14 (heat #C2220-1). Chemistry, fracture toughness data, and irradiation effects for this plate are contained in Section 4.

No identification of the weld metal was supplied and no chemistry or fracture toughness information are available.

The Monticello RPV material surveillance program is covered in Subsection 2.7 "Inspection and Testing," Chapter IV, of the Monticello FSAR.

6. PREDICTED ADJUSTED REFERENCE TEMPERATURE - IRRADIATION EFFECTS

Estimated maximum changes in RT_{NDT} and upper shelf fracture energy as a function of the end-of-life (EOL) fluence at the one-quarter thickness (1/4 T) depth of the vessel beltline materials are listed below. The updated predicted peak EOL fluence at the 1/4 T depth of the RPV beltline is 2.2×10^{18} n/cm^2 after 40 years of service. Transition temperature changes and changes in upper shelf energy were calculated in accordance with the rules of Regulatory Guide 1.99.

Results for the plate material are:

Limiting Plate Material (Cu = 0.17, P = 0.010%):	Heat #C2220-2
Transition Temperature Change (per Regulatory Guide 1.99):	66°F
Reference Temperature (Initial RT_{NDT}):	14°F
Adjusted Reference Temperature at EOL:	80°F

Since no copper and phosphorus contents are known for the weld filler material the upper limit curve in Regulatory Guide 1.99 must be used to estimate transition temperature changes due to irradiation. A shift of 155°F is shown at 2.2×10^{18} n/cm^2. This results in the predicted value of the adjusted reference temperature at EOL of 195°F (40°F + 155°F).

Since this predicted EOL reference temperature is less than 200°F, provision to permit thermal annealing of the RPV in accordance with 10 CFR 50 Appendix G is not required.