Wisconsin Electric
POWER COMPANY

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VPNPD-92-231
NRC-92-070

June 25, 1992

U. S. NUCLEAR REGULATORY COMMISSION
Document Control Desk
Mail Station P1-137
Washington, DC 20555

Gentlemen:

DOCKETS 50-266 AND 50-301
RESPONSE TO NRC GENERIC LETTER 92-01, REVISION 1
REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f)

NRC Generic Letter 92-01, "Reactor Vessel Structural Integrity," dated March 6, 1992, was issued to obtain information from licensees to enable the NRC to assess compliance with regulatory requirements and commitments regarding reactor vessel integrity. A response to Generic Letter 92-01 was requested within 120 days of the date of issuance. We understand that Generic Letter 92-01 was issued in view of certain concerns raised during NRC staff's review of reactor vessel integrity for the Yankee Nuclear Power Station.

The Babcock and Wilcox Owners Group's Reactor Vessel Working Group, under the direction of Wisconsin Electric and other member utilities, developed report BAW-2166, "B&W Owners Group Response to Generic Letter 92-01," which is enclosed. This report provides the information requested by Generic Letter 92-01 and was forwarded to the NRC by B&W Nuclear Service Company on June 17, 1992.

Generic Letter 92-01 presents the information requested in three sections (1, 2, and 3) which are further divided into a number of items. A tabular response format is used in BAW-2166 to respond to the individual sections and items contained in the Generic Letter. The response format is delineated in Chapters 3 and 6 of BAW-2166.

Wisconsin Electric sponsored and directed the development of BAW-2166 and has endorsed the data contained in BAW-2166 regarding our Point Beach Nuclear Plant. We believe the data contained in BAW-2166 regarding Point Beach Nuclear Plant satisfactorily responds to the information requested in Generic Letter 92-01. A summary of the data from that report, which is applicable to Point Beach, Units 1 and 2, is provided in the following paragraphs.

Add: McDONALD, D.
WICKMAN, K.
ELLIOT, B

A028
1/1

Section 1 of the Generic Letter requests information related to the licensee's surveillance program pursuant to Appendix H to 10 CFR Part 50. Section 1 is not applicable to Point Beach, Units 1 and 2 because they are currently part of an NRC-approved integrated surveillance program as listed in Enclosure 2 to the Generic Letter. Table 1 in the Point Beach, Units 1 and 2 chapters of BAW-2166 addresses the issue identified in Section 1.

Section 2 of Generic Letter 92-01 is divided into Items a and b. Item b contains a number of subitems - 1 through 6. Section 2, Item a discusses plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their current license period using the guidance of Regulatory Guide 1.99, Revision 2. Item a, asks addressees to provide upper shelf energy values for the limiting beltline weld and plate or forging. This data is provided in able 2 of BAW-2166 for the chapters applicable to PBNP Units 1 and 2. As noted in Table 2, both Point Beach units are projected to drop below 50 foot-pounds prior to the end of their current licensed life, using the guidance provided in Regulatory Guide 1.99 Revision 2. An analysis for our Point Beach Nuclear Plants to demonstrate adequate margins of safety to that required in ASME Section III Appendix G is scheduled to be performed in 1993 under the sponsorship of the B&W Owners Group Reactor Vessel Working Group. The Owners Group has completed analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants and have submitted the results to the NRC. The results of these analyses are anticipated to bound the outcome of both Point Beach units. Additionally, as previously reported in our Point Beach Nuclear Plant Unit 2 surveillance capsule S report dated October 15, 1991, correlations have been developed by the Owners Group for predicting the effects of neutron irradiation on Linde 80 Submerged Arc Welds. These results were reported in BAW-1803, Revision 1, "Correlations for Predicting the Effects of Neutron Irradiation on Linde 80 Submerged-Arc Welds," which was transmitted directly from B&W Nuclear Service Company to the NRC on October 4, 1991. This report demonstrates that for both Point Beach Nuclear Plant Unit 1 and Unit 2, the mean value of the upper shelf energy for the controlling weld metal will not decrease below 50 ft-lbs during the current 40-year license.

Section 2, Item b, requests licensees whose reactor vessels were constructed to an ASME code earlier than the Summer 1972 Agenda of the 1971 Edition to describe considerations given to certain material properties described in Subitems 1 through 6. As stated in Chapter 5 of BAW-2166, both Point Beach reactor vessels were constructed to the 1965 Edition. The answers to the questions of Subitems 1 through 6 along with the associated references are provided in Tables 2 through 7 of BAW-2166 chapters applicable to each Point Beach unit.

Section 3 requests licensees to provide information regarding commitments made to respond to Generic Letter 88-11. Section 3 is divided into Items a, b, and c.

Section 3, Item a, requests information regarding how the embrittlement effects of operating at a temperature below 525 °F were considered for Charpy upper shelf energy and reference temperature. This part is only applicable to Point Beach, Unit 1. Unit 1 was operated at reduced power and temperature for approximately four years because of steam generator concerns. Once the Unit 1 steam generators were replaced, the Unit was returned to normal power and temperature operation, as shown in Figure 4-4 of BAW-2166. As discussed in Chapter 4, Section 4.4 and Table 8 of BAW-2166, a surveillance capsule was removed both before and after the period of low power and low temperature operation. The results from these capsules show that the actual material behavior is conservatively estimated by Regulatory Guide 1.99, Revision 2. Therefore, due to this conservatism, this period of low temperature operation was not considered in determination of embrittlement effects.

Section 3, Item b, requests information regarding how surveillance results on the predicted amount of embrittlement were considered. Surveillance results from Point Beach Surveillance Program have not been used to predict the embrittlement of the Point Beach reactor vessels as stated in BAW-2166 Table 9 for both Point Beach units. In general, the predicted amounts of embrittlement have been determined by the generic methods outlined in regulatory guides and appropriate 10 CFR Part 50 regulations. Additionally, correlation techniques developed by the B&W Owners Group Reactor Vessel Working Group have been used.

Section 3, Item c, requests information regarding whether the measured shift in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if the measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Regulatory Guide 1.99, Revision 2. As depicted in BAW-2166 Table 10 for each Point Beach unit, no measured changes have exceeded these limits.

In addition to the enclosed BAW-2166 report, we have attached additional information which will contribute to your review of our reactor vessel integrity program. Attachment 1 provides a listing of our overall reactor vessel integrity program and Attachment 2 provides Unit 1 and 2 reactor vessel sketches of the beltline region material.

We believe this response has demonstrated our continued compliance with 10 CFR 50.60 and conformance to our commitments made in response to Generic Letter 88-11.

Please contact us should you have questions or require additional information regarding this response.

Sincerely,

Bob Link
Vice President
Nuclear Power

Enclosure (BAW-2166)

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector

Subscribed and sworn to before me
this 29th day of June, 1992.

Delores R Gryzkowski
Notary Public, State of Wisconsin

My Commission expires 5-22-94.

ATTACHMENT 1 POINT BEACH NUCLEAR PLANT REACTOR VESSEL INTEGRITY PROGRAM 1984 TO PRESENT	
PROJECT	**DATE COMPLETE**
1. Neutron exposure evaluation of Point Beach reactor vessels.	December 1984
2. Tested Unit 1 Surveillance Capsule T.	December 1984
3. 10 CFR 50.61 - Pressurized Thermal Shock (PTS) Submittal.	January 1986
Correction to PTS submittal.	March 1986
Safety evaluation report received from NRC.	July 1986
4. Reactor Vessel Life Extension Study.	
Initiated study in May 1986.	
Evaluation of fuel management techniques and internals modifications (shielding) to meet flux reduction goals.	September 1987
Identification of critical components in NSSS, including the reactor vessel and compilation of transient data associated with these components.	October 1987
Comprehensive scoping risk assessment to examine Point Beach specific concerns and the propriety of the flux reduction goals.	December 1987
Developed bases and specifications for a plantwide on-line fatigue monitoring system.	December 1987

5. Inservice Inspection	
a. Second <u>Unit 1</u> Reactor Vessel Ten-Year Exam.	
Performed ASME Code exam utilizing SwRI standard data acquisition system, including 50/70 tandem near surface search units.	May 1987
Performed exam using NES/Dynacon Ultrasonic Data Recording and Processing System (UDRPS) concurrent with ASME Code exam above.	May 1987
b. Second <u>Unit 2</u> Reactor Vessel Ten-year Exam.	
SWRI Enhanced Data Acquisition System (EDAS) was utilized.	October 1989
6. Joined Babcock and Wilcox Owner's Group (BWOG) Materials Committee.	August 1988
Full participant in BWOG Reactor Vessel Integrity Program (RVIP).	August 1988
Participant in BWOG Reactor Vessel Life Extension Surveillance Program (RVSP).	1989
Developing master integrated reactor vessel surveillance program to include Westinghouse utilities with Linde 80 welds in their reactor vessels. (BAW-1543)	March 1989
Submitted BAW-1543 Revision 3 to NRC.	October 1989
Safety evaluation report received from NRC for BAW-1543.	June 1991

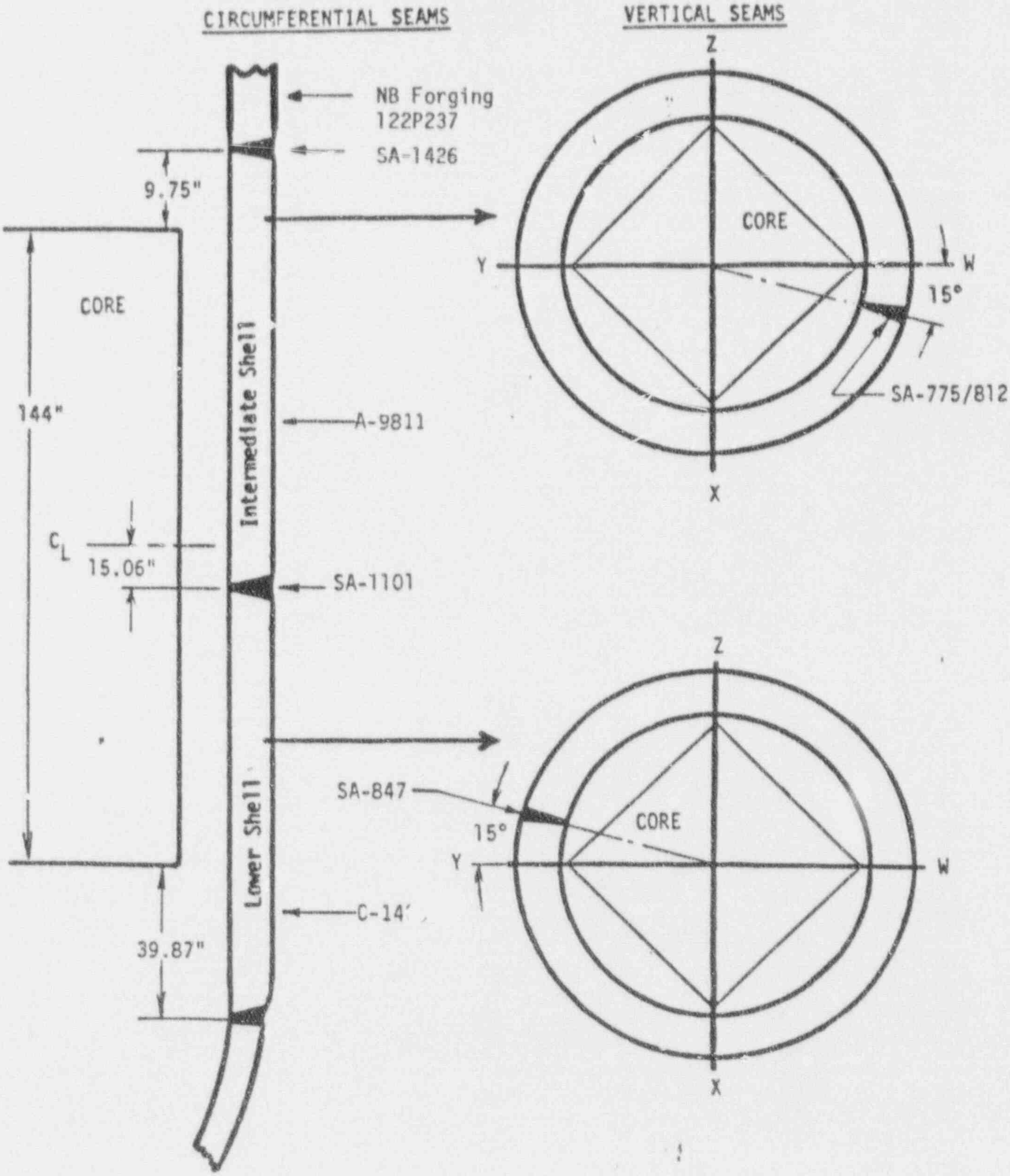
7. Installation of excore neutron dosimetry (radiometric monitors and solid state track recorders) over one octant of each unit's reactor vessel. Analysis of sensor sets and correlation of cavity measurements with transport calculations will be performed after each fuel cycle for first three sets. Thereafter, a three year interval will be used until sufficient data is obtained to increase the interval.	
Install mounting hardware and first set of dosimetry in Unit 2.	November 1988
Install mounting hardware and first set of dosimetry in Unit 1.	May 1989
First sensor set analyzed for Unit 2.	November 1990
First sensor set analyzed for Unit 1.	December 1990
Second sensor set analyzed for Unit 2.	October 1991
Second sensor set analyzed for Unit 1.	March 1992
8. Pilot project: On-line fatigue monitoring of Unit 2 pressurizer surge nozzle (related to reactor vessel life extension study fatigue evaluation).	November 1988
9. Implement super Low Leakage Loading Pattern (L4P) cores and axially-zoned hafnium inserts in the guide tubes of peripheral assemblies.	
Unit 1	May 1989
Unit 2	November 1989
10. Performed image enhancement of selected radiographs of important reactor coolant system components (reactor vessels, piping, steam generators, etc.) and retained radiograph image on media more permanent than original media.	1989

11. Submit revised heatup and cooldown curves using the guidance of Regulatory Guide 1.99, Revision 2. Technical Specification change approved by NRC.	August 1989 January 1990
12. Tested Unit 2 Surveillance Capsule S.	August 1991
13. Unit 1 & 2 Charpy Upper Shelf Energy Status and Unit 2 PTS Submittal	October 1991
14. Generic Letter 92-01, "Reactor Vessel Structural Integrity" Submittal	June 1992

Attachment 2

FIGURE 1

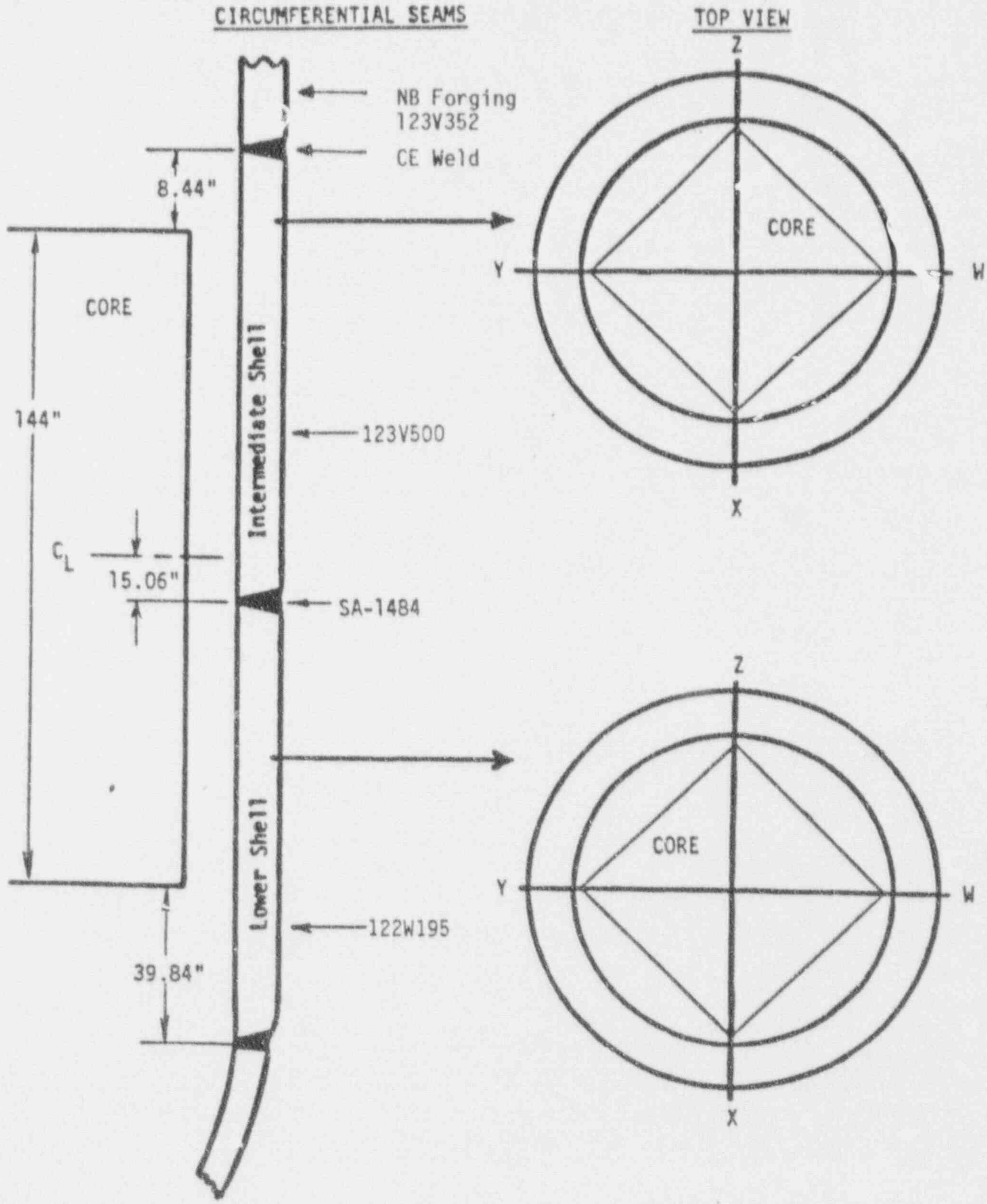
IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL FOR THE POINT BEACH UNIT NO. 1 REACTOR VESSEL



Attachment 2

FIGURE 2

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL FOR THE POINT BEACH UNIT NO. 2 REACTOR VESSEL



BAW-2166
JUNE 1992

THE B&W OWNERS GROUP

MATERIALS COMMITTEE

B&W OWNERS GROUP

RESPONSE TO GENERIC LETTER 92-01

B&W NUCLEAR SERVICE COMPANY

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BAW-2166
JUNE 1992

THE B&W OWNERS GROUP

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RESPONSE TO GENERIC LETTER 92-01

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<u>BAW-2166</u>

June 1992

B&W OWNERS GROUP
RESPONSE TO GENERIC LETTER 92-01

by

M. J. DeVan, L. B. Gross, and A. L. Lowe, Jr.

BWNS Document No. 77-2166-00
(See Section 7 for document signatures.)

Prepared for

<u>B&W Owners Group Reactor Vessel Working Group</u>
Commonwealth Edison Company
Duke Power Company
Entergy Operations, Inc.
Florida Power Corporation
Florida Power & Light Company
GPU Nuclear Corporation
Rochester Gas and Electric Corporation
Toledo Edison Company
Virginia Power Company
Wisconsin Electric Power Company

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1. INTRODUCTION

This report provides a response to the Nuclear Regulatory Commission (NRC) Generic Letter 92-01 for those nuclear power plants that are members of the B&W Owners Group Reactor Vessel Working Group.

Generic Letter 92-01, Revision 1, shown in Section 2 of this report, was issued by the NRC on March 6, 1992 and addressed to all holders of nuclear power plant operating licenses. The generic letter was issued to obtain information from the licensees to enable the NRC to assess the degree of compliance with regulatory requirements regarding reactor vessel integrity. Response is required within 120 days of the issue date; this comes to July 4, 1992. This document provides the required information, insofar as it is available, for the following plants:

Plant	Owner
Arkansas Nuclear One Unit 1	Entergy Operations, Inc.
Crystal River Unit 3	Florida Power Corporation
Davis-Besse Unit 1	Toledo Edison Company
R. E. Ginna Unit 1	Rochester Gas & Electric Corp.
Oconee Unit 1	Duke Power Company
Oconee Unit 2	Duke Power Company
Oconee Unit 3	Duke Power Company
Point Beach Unit 1	Wisconsin Electric Power Co.
Point Beach Unit 2	Wisconsin Electric Power Co.
Surry Unit 1	Virginia Electric & Power Co.
Surry Unit 2	Virginia Electric & Power Co.
Three Mile Island Unit 1	GPU Nuclear Corporation
Turkey Point Unit 3	Florida Power & Light Company
Turkey Point Unit 4	Florida Power & Light Company
Zion Unit 1	Commonwealth Edison Company
Zion Unit 2	Commonwealth Edison Company

2. GENERIC LETTER

Generic Letter 92-01, Revision 1, is shown below. (Enclosure 2 does not include Crystal River Unit 3; discussions with the NRC staff indicated that this is an inadvertent omission and that Crystal River Unit 3 is to be considered as if it is included in Enclosure 2.)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. [illegible]

March 6, 1992

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS (EXCEPT YANKEE ATOMIC ELECTRIC COMPANY, LICENSEE FOR THE YANKEE NUCLEAR POWER STATION)

SUBJECT: REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f) (GENERIC LETTER 92-01, REVISION 1)

This letter replaces Generic Letter 92-01 dated February 28, 1992. The background information concerning NRC's assessment of embrittlement in the Yankee Nuclear Power Station reactor vessel was drafted by staff some months ago and has now been clarified and updated to better reflect the licensee's extensive technical efforts regarding reactor vessel integrity. The section pertaining to required information has not changed.

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the staff's review of reactor vessel integrity for the Yankee Nuclear Power Station. In Section 50.60(a) of Title 10 of the Code of Federal Regulations (10 CFR 50.60(a)), the NRC requires that licensees for all light water nuclear power reactors meet fracture toughness requirements and have a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. In 10 CFR 50.60(b), where the requirements of Appendices G and H to 10 CFR Part 50 cannot be met, an exemption is necessary pursuant to 10 CFR 50.12. In 10 CFR 50.61 the NRC also provided fracture toughness requirements for protecting pressurized water reactors against pressurized thermal shock events. Licensees and permit holders have also made commitments in response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," to use the methodology in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The 10 CFR 50.60 and 10 CFR 50.61 requirements and GL 88-11 are in the overall regulatory program to maintain the structural integrity of the reactor vessel.

This generic letter is part of a program to evaluate reactor vessel integrity and take regulatory actions, if needed, to ensure that licensees and permit holders are complying with 10 CFR 50.60 and 10 CFR 50.61, and are fulfilling commitments made in response to GL 88-11. Enclosure 1 is a discussion of the applicable regulatory requirements. The NRC is requiring information on compliance under the provisions of 10 CFR 50.54(f).

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Assessment of Embrittlement for the Yankee Nuclear Power Station Reactor Vessel

In an effort to resolve concerns regarding the neutron embrittlement of the Yankee reactor vessel, the staff performed a safety assessment of the Yankee reactor vessel. The staff found that the licensee for the Yankee Nuclear Power Station might not be in compliance with 10 CFR 50.60 and 10 CFR 50.61.

The staff found that the Charpy upper shelf energy of the Yankee reactor vessel material could be as low as 35.5 foot-pounds which is less than the 50 foot-pound value required in Appendix G to 10 CFR Part 50. However, the licensee for the Yankee Nuclear Power Station had not performed the actions required in Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50. Since then, the licensee has performed an analysis in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 using criteria being developed by the American Society of Mechanical Engineers (ASME) to demonstrate margins of safety equivalent to those in the ASME Code.

The NRC expressed a concern regarding compliance with the requirements of Appendix H to 10 CFR Part 50. Section E 185 of the American Society for Testing and Materials (ASTM) Code requires that the licensee take sample specimens from actual material used in fabricating the beltline of the reactor vessel. These surveillance materials shall include one heat of base metal, one butt weld, and one weld "heat affected zone." The licensee for the Yankee Nuclear Power Station terminated the material surveillance program in 1965. Therefore, the Yankee Nuclear Power Station had no material surveillance program on July 26, 1983, when Appendix H to 10 CFR Part 50 became effective. Further, the samples irradiated at Yankee Rowe before 1965 were comprised only of base metal.

The licensee for the Yankee Nuclear Power Station had used the methodology in draft Regulatory Guide 1.99, Revision 2, to predict the effects of neutron embrittlement. The staff raised concerns regarding the licensee's application of the methodology. The specific issues were (1) the irradiation temperature, (2) the chemistry composition of reactor vessel material, and (3) the results of the material surveillance program.

The irradiation temperature at the Yankee Nuclear Power Station is between 454 °F and 520 °F, which is below the nominal irradiation temperature of 550 °F used in developing Regulatory Guide 1.99, Revision 2. A lower irradiation temperature increases the effect of neutron embrittlement. The regulatory guide indicates that for irradiation temperatures less than 525 °F, embrittlement effects should be considered to be greater than predicted by the methods of the guide. Adjustments that were made by the licensee were insufficient to account for this effect.

The results of the surveillance program from the Yankee Nuclear Power Station indicated that the increase in the reference temperature exceeds the mean-plus-two standard deviations as predicted by the procedures in Regulatory Guide 1.99, Revision 2. The regulatory guide states that the licensee should use credible surveillance data to predict the increase in reference temperature resulting from neutron irradiation.

The staff implemented RG 1.99, Revision 2, by issuing GL 88-11. In committing to GL 88-11, licensees have committed to calculate radiation embrittlement in accordance with the procedures documented in RG 1.99, Revision 2. To meet the limitations in Section 1.3 of the regulatory guide, the licensee should consider the effects on irradiation embrittlement during core critical operation with irradiation temperatures less than 525 °F. Section 2 of the regulatory guide states that the licensees should consider the effects of the results from its surveillance capsules.

The Summer 1972 Addenda of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code are the earliest code requirements for testing materials to determine their unirradiated reference temperature. The Yankee reactor vessel was constructed in 1959 to ASME Code, Section VIII. Therefore, the unirradiated reference temperature could not be established in accordance with the requirements of the Summer 1972 Addenda. The licensee for the Yankee Nuclear Power Station extrapolated the available test results to determine an unirradiated reference temperature. The staff determined that the licensee's extrapolation was not conservative.

The chemical composition of the Yankee reactor vessel welds is unknown. The material's sensitivity to neutron embrittlement depends on its chemical content. The licensee assumed that the chemistry of its welds was equivalent to that of the BR-3 reactor vessel in Mol, Belgium. The heat number of the wire used to fabricate the Yankee welds was not available. The licensee was assuming a chemical composition that was not based on its plant-specific information, since the chemical composition, in particular, the amount of copper, depends upon the heat number of the weld wire.

These factors prompted the staff to find that the licensee for the Yankee Nuclear Power Station had not fully considered plant-specific information in assessing compliance with 10 CFR 50.61. When plant-specific information is considered, the Yankee reactor vessel may have exceeded the screening criteria in 10 CFR 50.61.

Upon conducting the Yankee Nuclear Power Station review, the staff became concerned about other licensee's compliance with 10 CFR 50.60 and 10 CFR 50.61 and fulfillment of commitments made in response to GL 88-11. Thus, the staff is issuing this generic letter to obtain information to assess compliance with these regulations and fulfillment of commitments. The staff is continuing to pursue this concern with the Yankee Atomic Electric Company. Therefore, the Yankee Atomic Electric Company need not respond to this generic letter.

Required Information

Portions of the following information requested are not applicable to all addressees. The responses provided should, in these cases, indicate that the requested information is not applicable and why it is not applicable.

1. Certain addressees are requested to provide the following information regarding Appendix H to CFR Part 50:

   Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

2. Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

   a. Addressees of plants for which the Charpy upper shelf energy is predicted to be less then 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.

   b. Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

      (1) the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test;

      (2) the heat treatment received by all beltline and surveillance materials;

      (3) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;

(4) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;

(5) the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material; and

(6) the heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above.

3. Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

   a. How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525 °F were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.

   b. How their surveillance results on the predicted amount of embrittlement were considered.

   c. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.

Reporting Requirements

Pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), each addressee shall submit a letter within 120 days of the date of this generic letter providing the information described under "Required Information." The letter shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, under oath or affirmation. A copy shall also be submitted to the appropriate Regional Administrator. This generic letter requests information that will enable the NRC to verify that the licensee is complying with its current licensing basis regarding reactor vessel fracture toughness and material surveillance for the reactor coolant pressure boundary. Accordingly, an evaluation justifying this information request is not necessary under 10 CFR 50.54(f).

Backfit Discussion

This generic letter requests information that will enable the NRC staff to determine whether licensees are complying with their prior commitments and any license conditions regarding 10 CFR 50.60, 10 CFR 50.61, and GL 88-11. The staff is not establishing a new position for such compliance in this generic letter. The staff is requesting information to verify that the licensee is complying with its previously established commitments and is not establishing any new position. Therefore, this generic letter does not constitute a backfit and no documented evaluation or backfit analysis need be prepared.

Request for Voluntary Submittal of Impact Data

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires May 31, 1994. The estimated average number of burden hours is 200 person hours for each addressee's response, including the time required to assess the requirements, search data sources, gather and analyze the data, and prepare the required letters. This estimated average number of burden hours pertains only to the identified response-related matters and does not include the time to implement the actions required by the regulations. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to Ronald Minsk, Office of Information and Regulatory Affairs (3150-0011), NEOB-3019, Office of Management and Budget, Washington, DC 20503, and to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Division of Information Support Services, Office of Information and Resources Management, Washington, DC 20555.

Although no specific request or requirement is intended, the following information would assist the NRC in evaluating the cost of complying with this generic letter:

(1) the licensee staff's time and costs to perform requested inspections, corrective actions, and associated testing;

(2) the licensee staff's time and costs to prepare the requested reports and documentation;

(3) the additional short-term costs incurred to address the inspection findings such as the costs of the corrective actions or the costs of down time; and

(4) an estimate of the additional long-term costs that will be incurred as a result of implementing commitments such as the estimated costs of conducting future inspections or increased maintenance.

If you have any questions about this matter, please contact one of the NRC technical contacts or the lead project manager listed below.

Sincerely,

J G Partlow

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Applicable Regulatory Requirements
2. Plants with Integrated Programs
3. List of Recently Issued
   Generic Letters

Technical Contacts:
Barry J. Elliot, NRR
(301) 504-2709

Keith R. Wichman, NRR
(301) 504-2757

Lead Project Manager:
Daniel G. McDonald, NRR
(301) 504-1408

Enclosure 1

Regulatory Requirements Applicable to Reactor Vessel Structural Integrity

10 CFR 50.60

Pursuant to 10 CFR 50.60, all light water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to 10 CFR Part 50.

The fracture toughness of the reactor coolant pressure boundary required by 10 CFR 50.60 is necessary to provide adequate margins of safety during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. The material surveillance program required by 10 CFR 50.60 monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

Appendix G to 10 CFR Part 50 requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 50 ft-lb throughout the life of the vessel. Otherwise, licensees are required to provide demonstration of equivalent margins of safety in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 or perform actions in accordance with Paragraph V.C of Appendix G to 10 CFR Part 50.

Appendix H to 10 CFR Part 50 requires the surveillance program to meet the American Society for Testing and Materials (ASTM) Standard E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Further, Appendix H to 10 CFR Part 50 specifies the applicable edition of ASTM E 185. Appendix H to 10 CFR Part 50, as amended on July 26, 1983, requires that the part of the surveillance program conducted before the first capsule is withdrawn must meet the requirements of the 1973, the 1979, or the 1982 edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code under which the reactor vessel was purchased. The licensee may also use later editions of ASTM E 185 which have been endorsed by the NRC. The test procedures and reporting requirements for each capsule withdrawal after July 26, 1983 must meet the requirements of the 1982 edition of ASTM E 185 to the extent practical for the configuration of the specimens in the capsule. The licensee may use either the 1973, the 1979, or the 1982 edition of ASTM E 185 for each capsule withdrawal before July 26, 1983.

Licensees, especially those with reactor vessels purchased before ASTM issued the 1973 edition of ASTM E 185, may have surveillance programs that do not meet the requirements of Appendix H to 10 CFR Part 50 but may have alternative surveillance programs. The licensee may use these alternative surveillance programs in accordance with 10 CFR 50.60(b) if the licensee has been granted an exemption by the Commission under 10 CFR 50.12.

The licensee must monitor the test results from the material surveillance program. According to Paragraph III.C of Appendix H to 10 CFR Part 50, the results of the surveillance program may indicate that a technical specifications change is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits.

10 CFR 50.61

Pursuant to 10 CFR 50.61, there are fracture toughness requirements for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature. If the projected reference temperature exceeds the screening criteria established in 10 CFR 50.61, licensees are required to submit an analysis and schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, licensees shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed. In 10 CFR 50.61(b)(1), as amended effective June 14, 1991 (56 Fed Reg 22300 et. seq., May 15, 1991), licensees are required to submit their assessment by December 16, 1991, if the projected reference temperature will exceed the screening criteria before the expiration of the operating license.

Plant-specific information is required to be considered in assessing the level of neutron embrittlement as specified in 10 CFR 50.61(b)(3). This information includes but is not limited to the reactor vessel operating temperature and surveillance results.

Prediction of Irradiation Embrittlement

Paragraph V.A of Appendix G to 10 CFR Part 50 requires the prediction of the effects of neutron irradiation on reactor vessel materials. The extent of neutron embrittlement depends on the material properties, thermal environment, and results of the material surveillance program. In Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," the staff stated that it will use the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," in estimating the embrittlement of the materials in the reactor vessel beltline. All licensees and permittees have responded to Generic Letter 88-11 committing to use the methodology in Regulatory Guide 1.99, Revision 2, in predicting the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The methodology in Regulatory Guide 1.99, Revision 2, is also the basis in 10 CFR 50.61 in projecting the reference temperature.

Enclosure 2

Plants With Integrated Surveillance Programs Approved By The NRC

Oconee Units 1, 2, and 3
Arkansas Nuclear One Unit 1
Rancho Seco
Three Mile Island Unit 1
Davis-Besse
Ginna
Point Beach Units 1 and 2
Surry Units 1 and 2
Turkey Point Units 3 and 4
Zion Units 1 and 2

3. METHOD OF RESPONSE

3.1. Organization

The Generic Letter presents the information requests in three sections (1, 2, and 3) further divided into a number of items. Ten distinct sections/items were identified, each of which are presented in a table. The tables are identified as follows:

Table	GL 92-01 Reference	Subject
(1)	Section 1	10CFR50, Appendix H; Adherence to RVSP Requirements
(2)	Section 2, Item a	10CFR50, Appendix G; C_VUSE Requirements
(3)	Section 2, Item b, ¶ (1)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [unirradiated Charpy and RT_{NDT} values]
(4)	Section 2, Item b, ¶ (2)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [material heat treatment]
(5)	Section 2, Item b, ¶ (3)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [beltline material identification]
(6)	Section 2, Item b, ¶ (4)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [surveillance material identification]
(7)	Section 2, Item b, ¶ (5)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [chemical composition]

Table	GL 92-01 Reference	Subject
(8)	Section 3, Item a	Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
(9)	Section 3, Item b	Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
(10)	Section 3, Item c	Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Each of the above ten tables were prepared for each of the sixteen plants covered by this report. These tables are presented in Section 5 of this report.

3.2. Response Details

3.2.1. Abbreviations used in the response are as follows:

ART_{NDT}	Adjusted reference temperature
C_vUSE	Charpy upper-shelf energy
EOL	End of life
EST	Estimated value
NA	Not applicable
ND	Not determined
PTS	Pressurized thermal shock
RVSP	Reactor vessel surveillance program
RT_{NDT}	Reference temperature
ΔRT_{NDT}	Reference temperature shift
σ	Standard deviation

3.2.2. Material properties were determined at the ¼-thickness location, in accordance with 10CFR50, Appendix G, ¶ V.B, Footnote 2. Effects of neutron embrittlement were determined in accordance with the methods of

Regulatory Guide 1.99, Revision 2. The drop in C_vUSE was determined in accordance with Position 1 unless otherwise stated in the response tables. The end-of-life is taken as the time when 32 EFPY is achieved unless otherwise stated in the response tables.

4. IRRADIATION TEMPERATURE

Material sensitivity to irradiation embrittlement is directly affected by irradiation temperature. Over the temperature range that most light-water cooled reactors operate, the irradiation embrittlement is inversely related to irradiation temperature. However, since current generation pressurized water cooled reactors operate over a the relatively narrow temperature range (i.e. 529-556F RV inlet temperature), the relative sensitivity of the beltline materials as a function of temperature is easily overshadowed by other parameters such as variations in material properties and Charpy impact testing techniques. The development of Regulatory Guide 1.99, Revision 2, was based solely on surveillance data in the irradiation temperature range of 525 to 575F. Normally, the Regulatory Guide 1.99, Rev. 2 data is applied directly in the evaluation of a reactor vessel on the assumption that the reactor vessel temperature was always within this temperature range. However, as can be seen from a review of reactor coolant system temperature as a function of power, the inlet temperature can vary. This does not affect the monitoring of irradiation embrittlement of the reactor vessel because the surveillance capsules are located in the downcomer region of the reactor vessel and experience the same temperature history as the reactor vessel.

The reactor coolant system temperatures as a function of power for each plant included in this report are reviewed below. These data were provided by each plant owner and are as stated in their respective FSAR's.

4.1. B&W-Designed 177-FA Plants

Figure 4-1 shows the reactor vessel outlet temperature (T_{Hot}) and the reactor vessel inlet temperature (T_{Cold}) for the B&W 177-FA reactor vessels. This is representative of all 177-FA plants except Davis-Besse. These operating limits are characterized by a constant system average temperature and an increase in the inlet temperature (T_{Cold}) to 580F with a reduction in operating power. These temperature characteristics result from the fact that initial approach to power is controlled by the water level in the steam generator followed by a change in operation to maintain the system average temparature constant. The increase in inlet temperature may have the effect of minimizing irradiation embrittlement for these plants.

4.2. Davis-Besse

Figure 4-2 shows the reactor vessel outlet temperature (T_{Hot}) and the reactor vessel inlet temperature (T_{Cold}) for the Davis-Besse reactor vessel. The system behavior is similar to that of the other 177-FA plants with the exception that the change from level control to control of system average temperature is at approximately 28% power.

4.3. R. E. Ginna

Figure 4-3 shows the reactor vessel outlet temperature (T_{Hot}) and the reactor vessel inlet temperature (T_{Cold}) as a function of power for the R. E. Ginna reactor vessel. These operating limits are characterized by an increasing average temperature and a near constant reactor vessel inlet temperature for all power levels.

4.4. Point Beach Units 1 and 2

Figu.e 4-4 shows the reactor vessel outlet temperature (T_{Hot}) and the reactor vessel inlet temperature (T_{Cold}) as a function of power for the Point Beach Units 1 and 2 reactor vessels. These operating limits are characterized by an increasing average temperature and a small decrease in reactor vessel inlet temperature as power increase to 100%.

The Point Beach Unit 1 operated at a reduced power from approximately December 1, 1979 to October 1, 1983, as shown in Figure 4-4. During this period, the reactor vessel was operated at a temperature of 511F at 80% to 522F at 0% power.

This reduced operating temperature does not appear to have affected the irradiation embrittlement characteristics of the materials. Fortunately, a surveillance capsule was removed and evaluated prior to the reduced temperature operation. This capsule (Capsule R, WCAP-9357 and BAW-1803, Rev. 1) experienced a fluence of 2.10×10^{19} n/cm^2 (E > 1 MeV) and the weld metal exhibited an irradiation induced 30 ft-lb Charpy temperature shift of 165F. A similar capsule (Capsule T, WCAP-10736 and BAW-1803, Rev. 1) was removed and evaluated after the reduced temperature operation. The capsule experienced a fluence of 2.11×10^{19} n/cm^2 (E > 1 MeV) and the weld metal exhibited an irradiated induced Charpy 30 ft-lb temperature shift of 175F. Although it might be argued that this difference was caused by the reduced temperature exposure, the values are well within the expected scatter of Charpy impact test data. The comparable Regulatory Guide 1.99, Rev. 2 estimate for a fluence of 2.11×10^{19} n/cm^2, based on the weld metal chemical composition, is a shift of 196F. Consequently, the Regulatory Guide 1.99, Rev. 2 conservatively estimated the weld metal response to irradiation without the margin. Based on the Regulatory Guide 1.99, Revision 2, Position 2, and the data from four surveillance capsules estimates a shift value at 2.11×10^{19} n/cm^2 (E > 1 MeV) of 176F (without margin). Therefore, the reactor vessel material shift behavior as a result of exposure to irradiation is conservatively estimated by Regulatory Guide 1.99, Revision 2, both Position 1 and Position 2. Similar evaluation of the Charpy upper-shelf energy showed a value of 53 ft-lbs at a fluence of 2.10×10^{19} n/cm^2 (E > 1 MeV) for the capsule removed prior to the reduced temperature operation and a value of 55 ft-lbs at a fluence of 2.11×10^{19} n/cm^2 (E > 1 MeV) for the capsule removed after the reduced temperature operation. These values are well within the expected scatter of Charpy impact test data. The comparable Regulatory Guide 1.99, Revision 2 estimate for a fluence of 2.11×10^{19} n/cm^2 is an upper-shelf value of 37 ft-lbs. The Regulatory Guide 1.99, Revision 2 conservatively estimates the upper-shelf energy of the weld metal.

4.5. Surry Units 1 and 2

Figure 4-5 shows the reactor vessel outlet temperature (T_{Hot}) and the reactor vessel inlet temperature (T_{Cold}) as a function of power for the Surry Units 1 and 2 reactor vessels. These operating limits are characterized by an increasing average temperature and near constant reactor vessel inlet temperature for all power levels.

4.6. Turkey Point Units 3 and 4

Figure 4-6 shows the reactor vessel outlet temperature (T_{Hot}) and the reactor vessel inlet temperature (T_{Cold}) as a function of power for the Turkey Point Units 3 and 4 reactor vessels. These operating limits are characterized by an increasing average temperature and a near constant reactor vessel inlet temperature for all power levels.

4.7. Zion Units 1 and 2

Figure 4-7 shows the reactor vessel outlet temperature (T_{Hot}) and the reactor vessel inlet temperature (T_{Cold}) as a function of power for the Zion Units 1 and 2 reactor vessels. These operating limits are characterized by an increasing average temperature with increasing power levels. The inlet temperature decreases with increasing power and reaches a minimum at 100% power.

Figure 4-1. Reactor Coolant System Temperatures as a Function of Power for B&W 177-FA Plants Except Davis-Besse

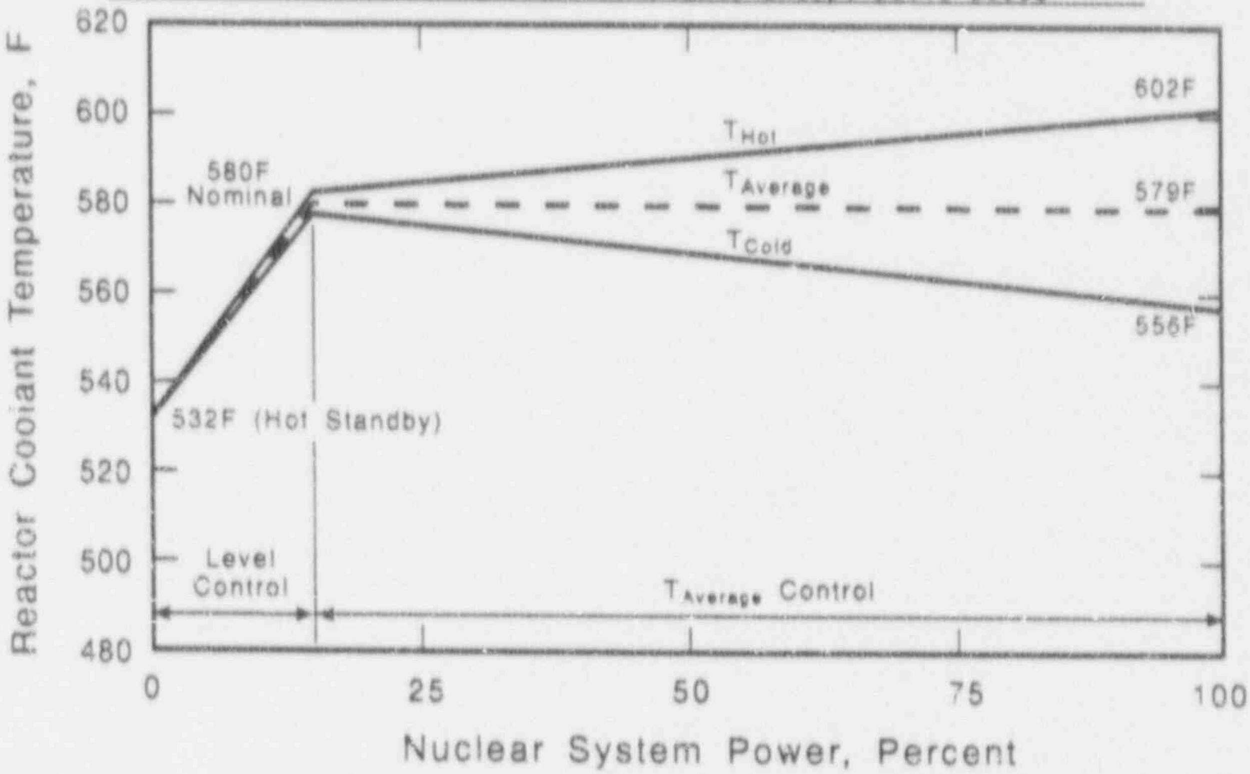


Figure 4-2. Reactor Coolant System Temperatures as a Function of Power for Davis-Besse

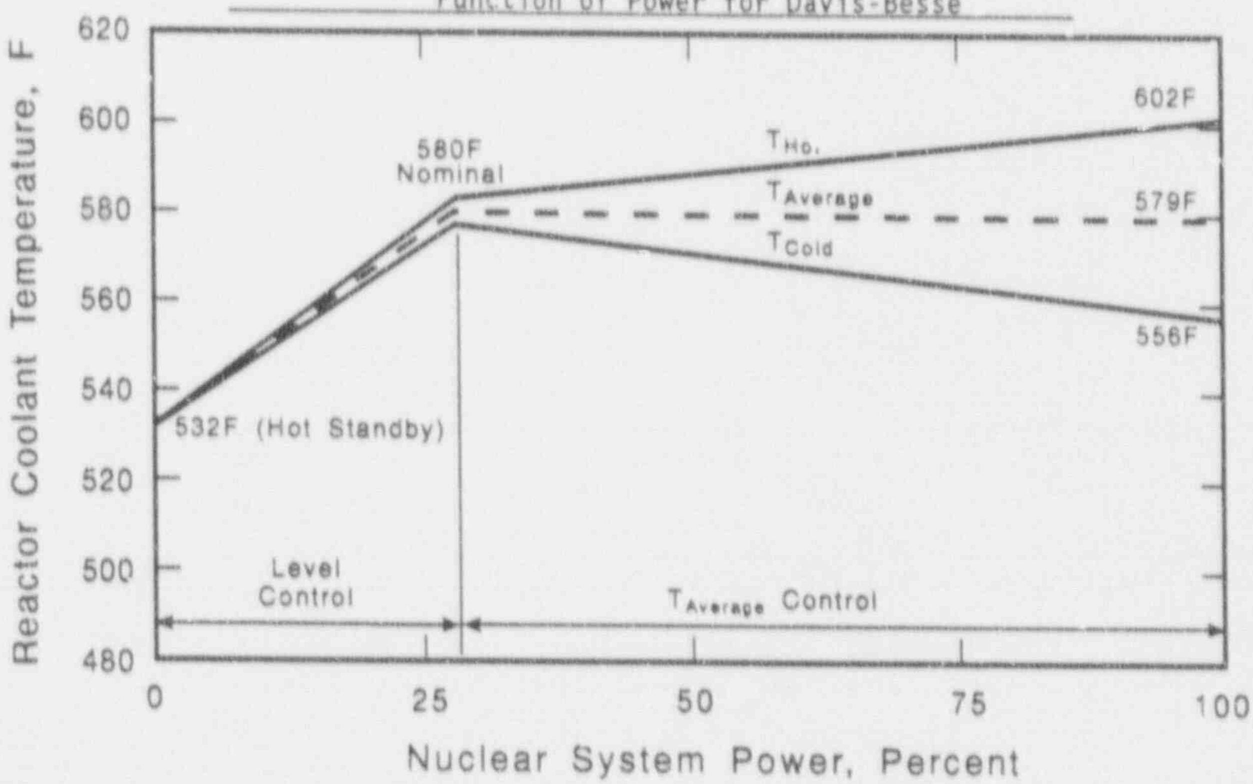


Figure 4-3. Reactor Coolant System Temperatures as a Function of Power for Westinghouse-Designed Plants for R. E. Ginna Unit 1

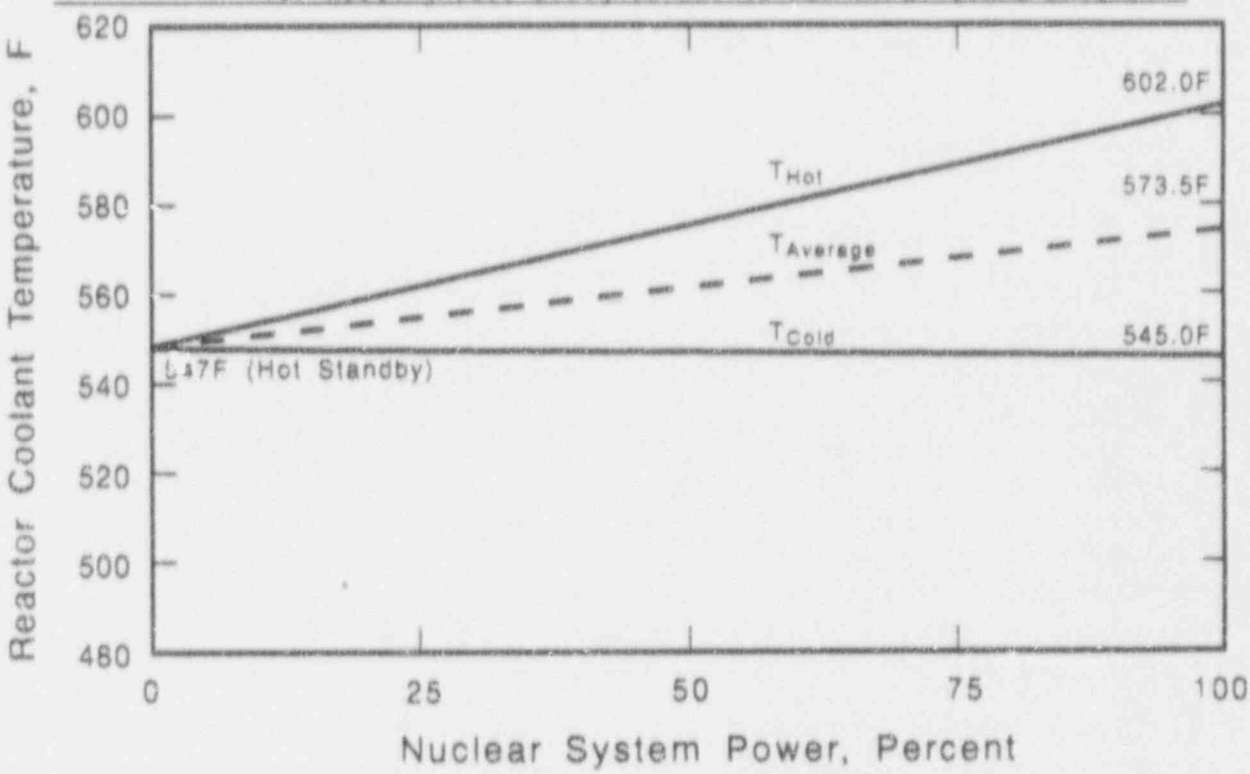


Figure 4-4. Reactor Coolant System Temperatures as a Function of Power for Westinghouse-Designed Plants for Point Beach Units 1 and 2

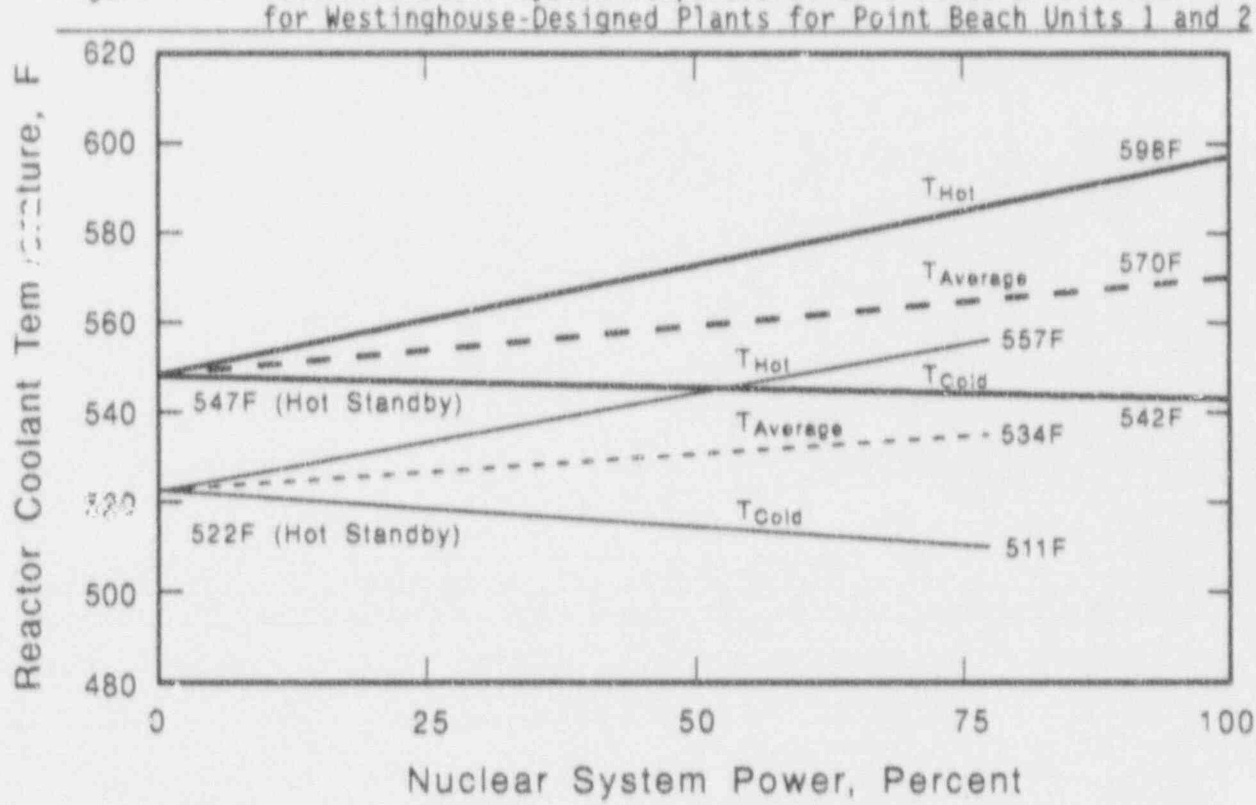


Figure 4-5. Reactor Coolant System Temperatures as a Function of Power for Westinghouse-Designed Plants for Surry Units 1 and 2

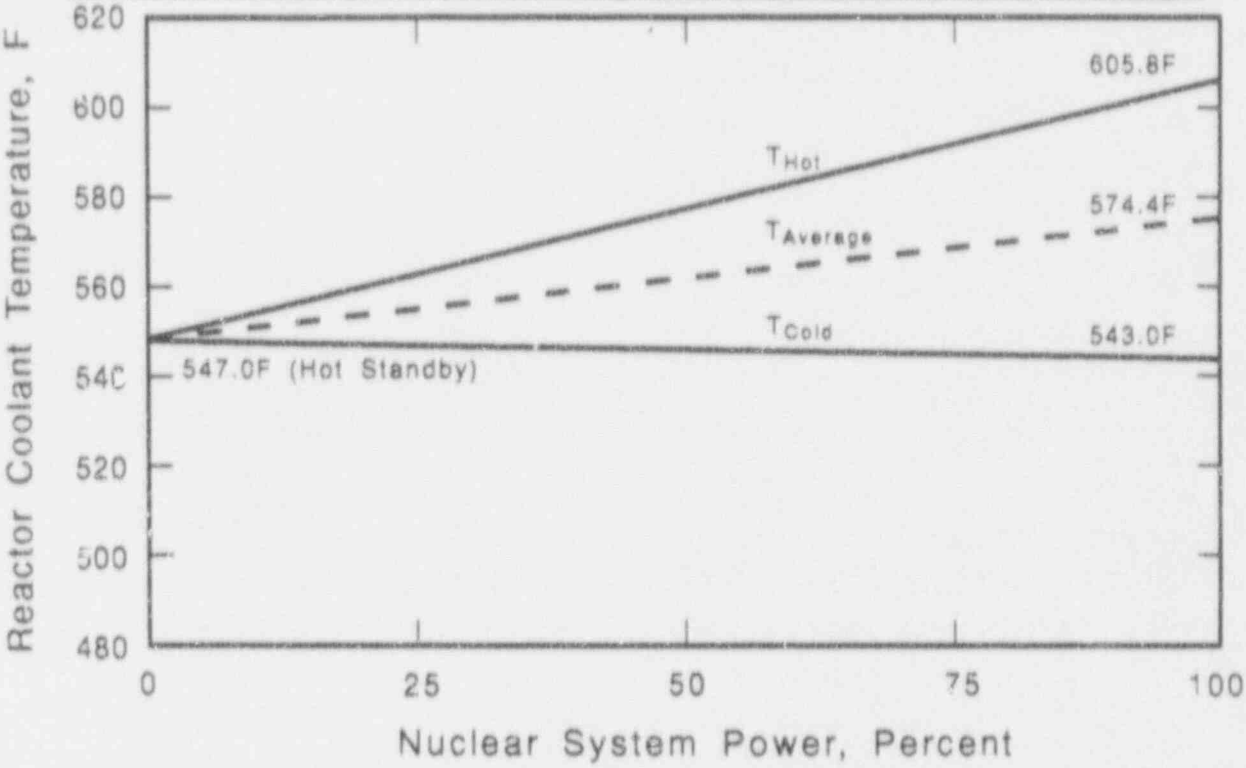


Figure 4-6. Reactor Coolant System Temperatures as a Function of Power for Westinghouse-Designed Plants for Turkey Point Units 3 and 4

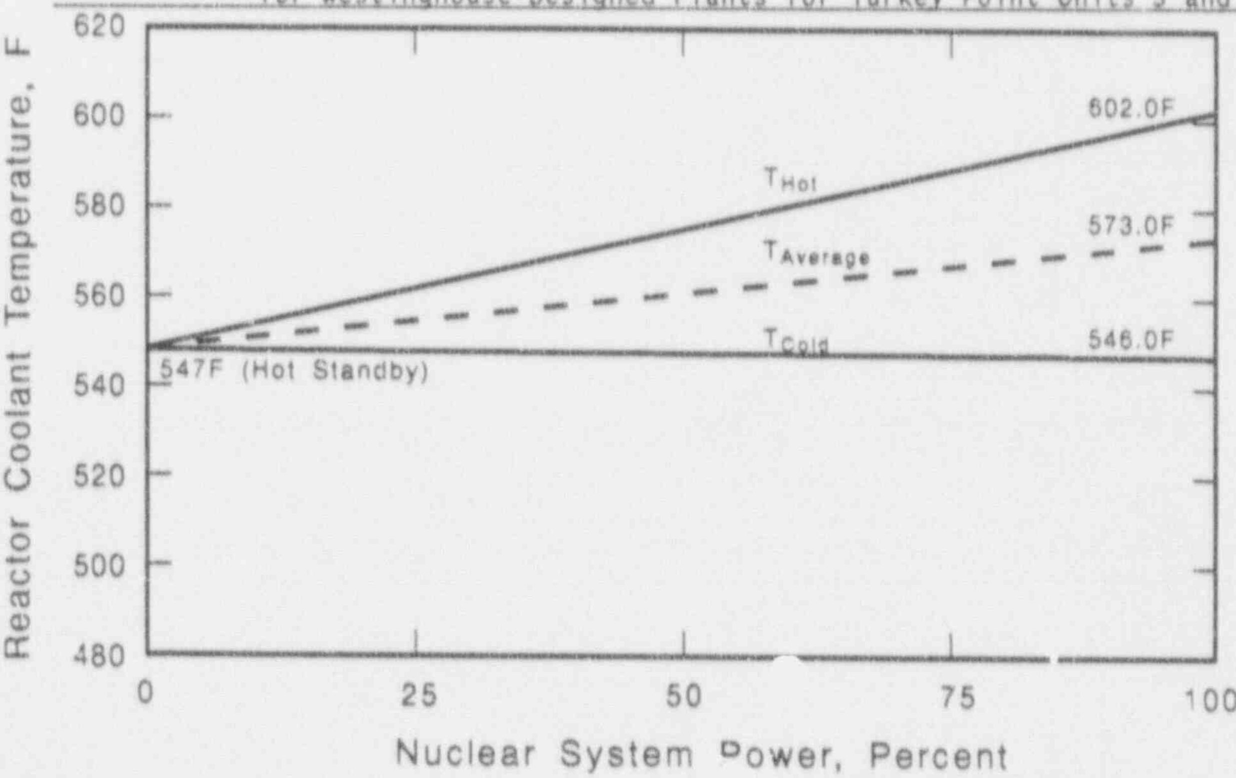
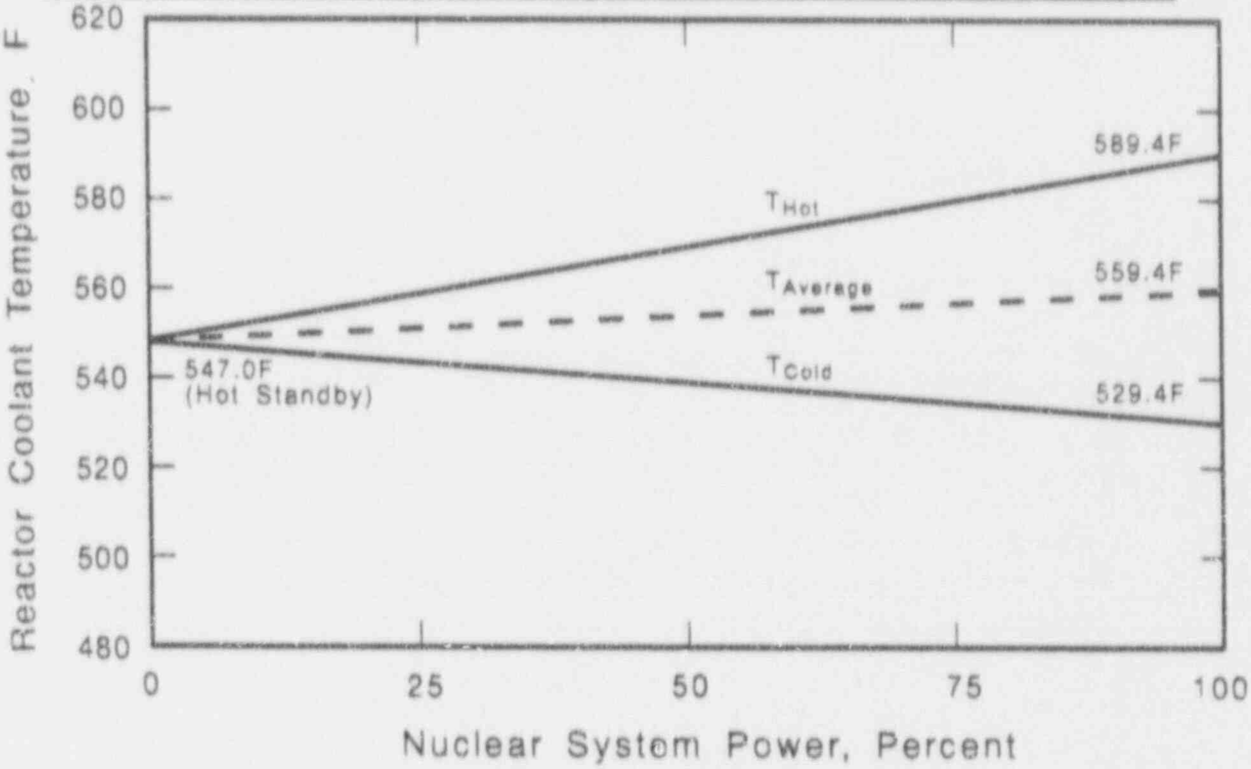


Figure 4-7. Reactor Coolant System Temperatures as a Function of Power for Westinghouse-Designed Plants for Zion Units 1 and 2



5. SUPPLEMENTARY INFORMATION

5.1. Construction Code

The reactor vessels for the following plants were fabricated in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The Edition and Addenda (where applicable) of the Code are noted.

Plant	Section III Edition and Addenda
Arkansas Nuclear One Unit 1	1965 Edition, Summer 1967 Addenda
Crystal River Unit 3	1965 Edition, Summer 1967 Addenda
Davis-Besse Unit 1	1968 Edition, Summer 1968 Addenda
R. E. Ginna Unit 1	1965 Edition
Oconee Unit 1	1965 Edition, Summer 1967 Addenda
Oconee Unit 2	1965 Edition, Summer 1967 Addenda
Oconee Unit 3	1965 Edition, Summer 1967 Addenda
Point Beach Unit 1	1965 Edition
Point Beach Unit 2	1965 Edition
Surry Unit 1	Not available, final assembly by Rotterdam
Surry Unit 2	Not available, final assembly by Rotterdam
Three Mile Island Unit 1	1965 Edition, Summer 1967 Addenda
Turkey Point Unit 3	1965 Edition, Summer 1966 Addenda
Turkey Point Unit 4	1965 Edition, Summer 1966 Addenda
Zion Unit 1	1965 Edition, Summer 1966 Addenda
Zion Unit 2	1965 Edition, Summer 1966 Addenda

5.2. Fluence Predictions

Peak fluence predictions for the beltline materials for each plant are presented in Table 5.2-1.

Table 5.2-1. Fluence Predictions for Beltline Region Materials

Arkansas Nuclear One Unit 1

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
AYN 131	Lower Nozzle Belt Forging	3.19E+18	1.92E+18	8.62E+18	5.18E+18
C5120-2	Upper Shell Plate	3.63E+18	2.18E+18	9.79E+18	5.88E+18
C5114-2	Upper Shell Plate	3.63E+18	2.18E+18	9.79E+18	5.88E+18
C5120-1	Lower Shell Plate	3.48E+18	2.09E+18	9.40E+18	5.64E+18
C5114-1	Lower Shell Plate	3.48E+18	2.09E+18	9.40E+18	5.64E+18
WF-182-1	Nozzle Belt to Upper Shell Circ. Weld	3.19E+18	1.92E+18	8.62E+18	5.18E+18
WF-112	Upper Shell to Lower Shell Circ. Weld	3.48E+18	2.09E+18	[illegible]	5.64E+18
SA-1788	Lower Shell to Dutchman Circ. Weld	2.03E+16	1.22E+16	5.48E+16	3.29E+16
WF-18	Upper Shell Longit. Weld	2.61E+18	1.57E+18	7.05E+18	4.23E+18
WF-18	Lower Shell Longit. Weld	2.58E+18	1.55E+18	6.95E+18	4.17E+18

Crystal River Unit 3

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
AJZ 94	Lower Nozzle Belt Forging	2.39E+18	1.44E+18	7.53E+18	4.52E+18

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Crystal River Unit 3 (Cont.)

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
C4344-1	Upper Shell Plate	2.72E+18	1.63E+18	8.56E+18	5.14E+18
C4344-2	Upper Shell Plate	2.72E+18	1.63E+18	8.56E+18	5.14E+18
C4347-1	Lower Shell Plate	2.61E+18	1.57E+18	8.22E+18	4.94E+18
C4347-2	Lower Shell Plate	2.61E+18	1.57E+18	8.22E+18	4.94E+18
SA-1769	Nozzle Belt to Upper Shell Circ. Weld (40% ID)	2.39E+18	1.44E+18	7.53E+18	4.52E+18
WF-169-1	Nozzle Belt to Upper Shell Circ. Weld (60% OD)	---	---	---	---
WF-70	Upper Shell to Lower Shell Circ. Weld	2.61E+18	1.57E+18	8.22E+18	4.94E+18
WF-154	Lower Shell to Dutchman Circ. Weld	1.52E+16	9.15E+15	4.79E+16	2.88E+16
WF-18	Upper Shell Longit. Weld	2.53E+18	1.52E+18	7.96E+18	4.78E+18
WF-8	Upper Shell Longit. Weld	2.53E+18	1.52E+18	7.96E+18	4.78E+18
SA-1580	Lower Shell Longit. Weld	2.22E+18	1.33E+18	6.98E+18	4.19E+18

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Davis-Besse Unit 1

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
ADB 203	Nozzle Belt Forging	3.92E+17	2.35E+17	1.50E+18	9.01E+17
AKJ 233	Upper Shell Forging	2.80E+18	1.68E+18	1.07E+19	6.43E+18
BCC 241	Lower Shell Forging	2.80E+18	1.68E+18	1.07E+19	6.43E+18
WF-232	Nozzle Belt to Upper Shell Circ. Weld (9% ID)	3.92E+17	---	1.50E+18	---
WF-233	Nozzle Belt to Upper Shell Circ. Weld (91% OD)	---	2.35E+17	---	9.01E+17
WF-182-1	Upper Shell to Lower Shell Circ. Weld	2.80E+18	1.68E+18	1.07E+19	6.43E+18
WF-232	Lower Shell to Dutchman Circ. Weld (12% ID)	1.57E+16	---	6.00E+16	---
WF-233	Lower Shell to Dutchman Circ. Weld (88% OD)	---	9.42E+15	---	3.60E+16

R. E. Ginna Unit 1

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
123P118VA1	Nozzle Belt Belt Forging	2.05E+18	1.39E+18	3.69E+18	2.50E+18
125S255VA1	Interm. Shell Forging	1.86E+19	1.26E+19	3.35E+19	2.27E+19

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

R. E. Ginna Unit 1 (Cont.)

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
125P666VA1	Lower Shell Forging	1.86E+19	1.26E+19	3.35E+19	2.27E+19
SA-1101	Nozzle Belt to Interm. Shell Circ. Weld	2.05E+18	1.39E+18	3.72E+18	2.52E+18
SA-847	Interm. Shell to Lower Shell Circ. Weld	1.86E+19	1.26E+19	3.35E+19	2.27E+19
SA-848	Lower Shell to Dutchman Circ. Weld	NA	NA	NA	NA

Oconee Unit 1

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
AHR 54	Lower Nozzle Belt Forging	6.20E+17	3.72E+17	1.18E+18	7.09E+17
[illegible]	Interm. Shell Plate	4.20E+18	2.52E+18	7.96E+18	4.78E+18
[illegible]	Upper Shell Plate	4.77E+18	2.86E+18	9.04E+18	5.43E+18
[illegible]	Upper Shell Plate	4.77E+18	2.86E+18	9.04E+18	5.43E+18
C2800-1	Lower Shell Plate	4.58E+18	2.75E+18	8.68E+18	5.21E+18
C2800-2	Lower Shell Plate	4.58E+18	2.75E+18	8.68E+18	5.21E+18
SA-1135	Nozzle Belt to Interm. Shell Circ. Weld	6.20E+17	3.72E+17	1.18E+18	7.09E+17

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Oconee Unit 1 (Cont.)

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
SA-1229	Interm. Shell to Upper Shell Circ. Weld (61% ID)	4.20E+18	2.52E+18	7.96E+18	4.78E+18
WF-25	Interm. Shell to Upper Shell Circ. Weld (39% OD)	---	---	---	---
SA-1585	Upper Shell to Lower Shell Circ. Weld	4.58E+18	2.75E+18	8.68E+18	5.21E+18
WF-9	Lower Shell to Dutchman Circ. Weld	2.67E+16	1.60E+16	5.06E+16	3.04E+16
SA-1073	Interm. Shell Longit. Weld	3.32E+18	1.99E+18	6.28E+18	3.77E+18
SA-1493	Upper Shell Longit. Weld	3.82E+18	2.29E+18	7.23E+18	4.34E+18
SA-1426	Lower Shell Longit. Weld	3.85E+18	2.31E+18	7.29E+18	4.38E+18
SA-1430	Lower Shell Longit. Weld	3.85E+18	2.31E+18	7.29E+18	4.38E+18

Oconee Unit 2

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
AMX 77	Lower Nozzle Belt Forging	3.88E+18	2.33E+18	8.42E+18	5.06E+18
AAW 163	Upper Shell Forging	4.41E+18	2.65E+18	9.57E+18	5.75E+18

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Oconee Unit 2 (Cont.)

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
AWG 164	Lower Shell Forging	4.23E+18	2.54E+18	9.19E+18	5.52E+18
WF-154	Nozzle Belt to Upper Shell Circ. Weld	3.88E+18	2.33E+18	8.42E+18	5.06E+18
WF-25	Upper Shell to Lower Shell Circ. Weld	4.23E+18	2.54E+18	9.19E+18	5.52E+18
WF-112	Lower Shell to Dutchman Circ. Weld	2.47E+16	1.48E+16	5.36E+16	3.22E+16

Oconee Unit 3

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
4680	Lower Nozzle Belt Forging	3.85E+18	2.31E+18	8.26E+18	4.96E+18
AWS 192	Upper Shell Forging	4.37E+18	2.62E+18	9.39E+18	5.64E+18
ANK 191	Lower Shell Forging	4.20E+18	2.52E+18	9.01E+18	5.41E+18
WF-200	Nozzle Belt to Upper Shell Circ. Weld	3.85E+18	2.31E+18	8.26E+18	4.96E+18
WF-67	Upper Shell to Lower Shell Circ. Weld (75% ID)	4.20E+18	2.52E+18	9.01E+18	5.41E+18

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Oconee Unit 3 (Cont.)

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
WF-70	Upper Shell to Lower Shell Circ. Weld (25% OD)	---	---	---	---
WF-169-1	Lower Shell to Dutchman Circ. Weld	2.45E+16	1.47E+16	5.26E+16	3.16E+16

Point Beach Unit 1

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
122P237VA1	Nozzle Belt Forging	1.71E+18	1.16E+18	2.95E+18	2.00E+18
A9811-1	Interm. Shell Plate	1.55E+19	1.05E+19	2.68E+19	1.81E+19
C1423-1	Lower Shell Plate	1.52E+19	1.03E+19	2.33E+19	1.58E+19
SA-1426	Nozzle Belt to Interm. Shell Circ. Weld	1.71E+18	1.16E+18	2.95E+18	2.00E+18
SA-1101	Interm. Shell to Lower Shell Circ. Weld	1.52E+19	1.03E+19	2.33E+19	1.58E+19
SA-1101	Lower Shell to Dutchman Circ. Weld	---	---	---	---

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Point Beach Unit 1 (Cont.)

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
SA-812	Interm. Shell Longit. Weld (27% ID)	9.64E+18	6.53E+18	1.71E+19	1.16E+19
SA-775	Interm. Shell Longit. Weld (73% OD)	---	---	---	---
SA-847	Lower Shell Longit. Weld	9.53E+18	6.45E+18	1.56E+19	1.06E+19

Point Beach Unit 2

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
Not avail.	Nozzle Belt Forging	2.00E+18	1.35E+18	3.50E+18	2.37E+18
123V500VA1	Interm. Shell Forging	1.67E+19	1.13E+19	2.92E+19	1.98E+19
122W195VA1	Lower Shell Forging	1.65E+19	1.12E+19	2.66E+19	1.80E+19
Not avail.	Nozzle Belt to Interm. Shell Circ. Weld	2.00E+18	1.35E+18	3.50E+18	2.37E+18
SA-1484	Interm. Shell to Lower Shell Circ. Weld	1.64E+19	1.11E+19	2.56E+19	1.73E+19
Not. avail.	Lower Shell to Dutchman Circ. Weld	---	---	---	---

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Surry Unit 1

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
122V109VA1	Nozzle Belt Forging	2.48E+18	1.56E+18	5.27E+18	3.31E+18
C4326-1	Interm. Shell Plate	2.07E+19	1.30E+19	4.39E+19	2.76E+19
C4326-2	Interm. Shell Plate	2.07E+19	1.30E+19	4.39E+19	2.76E+19
C4415-1	Lower Shell Plate	2.07E+19	1.30E+19	4.39E+19	2.76E+19
C4415-1	Lower Shell Plate	2.07E+19	1.30E+19	4.39E+19	2.76E+19
J726	Nozzle Belt to Interm. Shell Circ. Weld	2.48E+18	1.56E+18	5.27E+18	3.31E+18
SA-1585	Interm. Shell to Lower Shell Circ. Weld (40% ID)	2.07E+19	1.30E+19	4.39E+19	2.76E+19
SA-1650	Interm. Shell to Lower Shell Circ. Weld (60% OD)	---	---	---	---
SA-1494	Interm. Shell Longit. Weld	3.34E+18	2.10E+18	7.08E+18	4.45E+18
SA-1494	Lower Shell to Longit. Weld	3.34E+18	2.10E+18	7.08E+18	4.45E+18
SA-1526	Lower Shell Longit. Weld	3.34E+18	2.10E+18	7.08E+18	4.45E+18

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Surry Unit 2

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
123V303VA1	Nozzle Belt Forging	2.48E+18	1.56E+18	4.45E+18	2.80E+18
C4208-2	Interm. Shell Plate	2.07E+19	1.30E+19	3.71E+19	2.33E+19
C4339-1	Interm. Shell Plate	2.07E+19	1.30E+19	3.71E+19	2.33E+19
C4331-1	Lower Shell Plate	2.07E+19	1.30E+19	3.71E+19	2.33E+19
C4339-2	Lower Shell Plate	2.07E+19	1.30E+19	3.71E+19	2.33E+19
L737	Nozzle Belt to Interm. Shell Circ. Weld	2.48E+18	1.56E+18	4.45E+18	2.80E+18
R3008	Interm. Shell to Lower Shell Circ. Weld	2.07E+19	1.30E+19	3.71E+19	2.33E+19
SA-1585	Interm. Shell Longit. Weld	4.32E+18	2.71E+18	7.75E+18	4.87E+18
SA-1585	Interm. Shell Longit. Weld (50% ID)	4.32E+18	2.71E+18	7.75E+18	4.87E+18
WF-4	Interm. Shell Longit. Weld (50% OD)	---	---	---	---
WF-4	Lower Shell Longit. Weld	4.32E+18	2.71E+18	7.75E+18	4.87E+18
WF-4	Lower Shell Longit. Weld (63% ID)	4.32E+18	2.71E+18	7.75E+18	4.87E+18

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Surry Unit 2 (Cont.)

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
WF-4	Interm. Shell Longit. Weld (37% OD)	---	---	---	---

Three Mile Island Unit 1

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 26.17 EFPY IS	Fluence, 26.17 EFPY T/4
ARY 59	Lower Nozzle Belt Forging	2.68E+18	1.61E+18	6.60E+18	3.96E+18
C2789-1	Upper Shell Plate	3.04E+18	1.83E+18	7.50E+18	4.50E+18
C2789-2	Upper Shell Plate	3.04E+18	1.83E+18	7.50E+18	4.50E+18
C3307-1	Lower Shell Plate	2.92E+18	1.75E+18	7.20E+18	4.32E+18
C3251-1	Lower Shell Plate	2.92E+18	1.75E+18	7.20E+18	4.32E+18
WF-70	Nozzle Belt to Interm. Shell Circ. Weld	2.68E+18	1.61E+18	6.60E+18	3.96E+18
WF-25	Upper Shell to to Lower Shell Circ. Weld	2.92E+18	1.75E+18	7.20E+18	4.32E+18
WF-67	Lower Shell to Dutchman Circ. Weld (50% ID)	1.70E+16	1.02E+16	4.20E+16	2.52E+16
WF-70	Lower Shell to Dutchman Circ. Weld (50% OD)	---	---	---	---

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Three Mile Island Unit 1 (Cont.)

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 26.17 EFPY IS	Fluence, 26.17 EFPY T/4
WF-8	Upper Shell Longit. Weld	3.04E+18	1.83E+18	7.50E+18	4.50E+18
SA-1526	Lower Shell Longit. Weld	2.69E+18	1.62E+18	6.50E+18	3.90E+18
SA-1526	Lower Shell Longit. Weld (37% ID)	2.69E+18	1.62E+18	6.50E+18	3.90E+18
SA-1494	Lower Shell Longit. Weld (63% OD)	---	---	---	---

Turkey Point Unit 3

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
122S146VA1	Nozzle Belt Forging	1.80E+18	1.13E+18	3.17E+18	1.99E+18
123P461VA1	Interm. Shell Forging	1.50E+19	9.42E+18	2.64E+19	1.66E+19
123S266VA1	Lower Shell Forging	1.50E+19	9.42E+18	2.64E+19	1.66E+19
SA-1484	Nozzle Belt to Interm. Shell Circ. Weld	1.80E+18	1.13E+18	3.17E+18	1.99E+18
SA-1101	Interm. Shell to Lower Shell Circ. Weld	1.50E+19	9.42E+18	2.64E+19	1.66E+19
SA-1135	Lower Shell to Dutchman Circ. Weld	---	---	---	---

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Turkey Point Unit 4

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
124S309VA1	Nozzle Belt Forging	1.64E+18	1.03E+18	3.04E+18	1.91E+18
123P481VA1	Interm. Shell Forging	1.37E+19	8.60E+18	2.53E+19	1.59E+19
122S180VA1	Lower Shell Forging	1.37E+19	8.60E+18	2.53E+19	1.59E+19
WF-67	Nozzle Belt to Interm. Shell Circ. Weld (67% ID)	1.64E+18	1.03E+18	3.04E+18	1.91E+18
WF-70	Nozzle Belt to Interm. Shell Circ. Weld (33% OD)	---	---	---	---
SA-1101	Interm. Shell to Lower Shell Circ. Weld	1.37E+19	8.60E+18	2.53E+19	1.59E+19
SA-1135	Lower Shell to Dutchman Circ. Weld	---	---	---	---

Zion Unit 1

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
ANA 102	Lower Nozzle Belt Forging	3.22E+18	1.93E+18	8.65E+18	5.19E+18
C3795-2	Interm. Shell Plate	6.44E+18	3.87E+18	1.73E+19	1.04E+19
B7835-1	Interm. Shell Plate	6.44E+18	3.87E+18	1.73E+19	1.04E+19

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Zion Unit 1 (Cont.)

Material	Location	Fluence, 12/16/91		Fluence, 32 EFPY	
		IS	T/4	IS	T/4
C3799-2	Lower Shell Plate	6.44E+18	3.87E+18	1.73E+19	1.04E+19
B7823-1	Lower Shell Plate	6.44E+18	3.87E+18	1.73E+19	1.04E+19
WF-154	Nozzle Belt to Interm. Shell Circ. Weld (82% ID)	3.22E+18	1.93E+18	8.65E+18	5.19E+18
SA-1769	Nozzle Belt to Interm. Shell Circ. Weld (18% OD)	---	---	---	---
WF-70	Interm. Shell to Lower Shell Circ. Weld	6.44E+18	3.87E+18	1.73E+19	1.04E+19
WF-154	Lower Shell to Dutchman Circ. Weld	---	---	---	---
WF-4	Interm. Shell Longit. Weld	2.34E+18	1.41E+18	6.29E+18	3.78E+18
WF-8	Interm. Shell Longit. Weld (39% ID)	2.34E+18	1.41E+18	6.29E+18	3.78E+18
WF-4	Interm. Shell Longit. Weld (61% OD)	---	---	---	---
WF-8	Lower Shell Longit. Weld	2.34E+18	1.41E+18	6.29E+18	3.78E+18

Table 5.2-1. Fluence Predictions for Beltline Region Materials (Cont.)

Zion Unit 2

Material	Location	Fluence, 12/16/91 IS	Fluence, 12/16/91 T/4	Fluence, 32 EFPY IS	Fluence, 32 EFPY T/4
ZV 3855	Lower Nozzle Belt Forging	3.22E+18	1.93E+18	8.45E+18	5.07E+18
B8006-1	Interm. Shell Plate	6.43E+18	3.86E+18	1.69E+19	1.01E+19
B8040-1	Interm. Shell Plate	6.43E+18	3.86E+18	1.69E+19	1.01E+19
C4007-1	Lower Shell Plate	6.43E+18	3.86E+18	1.69E+19	1.01E+19
B8029-1	Lower Shell Plate	6.43E+18	3.86E+18	1.69E+19	1.01E+19
WF-200	Nozzle Belt to Interm. Shell Circ. Weld	3.22E+18	1.93E+18	8.45E+18	5.07E+18
SA-1769	Interm. Shell to Lower Shell Circ. Weld	6.43E+18	3.86E+18	1.69E+19	1.01E+19
WF-154	Lower Shell to Dutchman Circ. Weld	---	---	---	---
WF-70	Interm. Shell Longit. Weld	2.30E+18	1.38E+18	6.04E+18	3.63E+18
WF-29	Lower Shell Longit. Weld	2.30E+18	1.38E+18	6.04E+18	3.63E+18

6. RESPONSE TO GENERIC LETTER 92-01

The following tables are submitted in response to the information requested in Generic Letter 92-01.

<u>Arkansas Nuclear One Unit 1</u>

<u>Crystal River Unit 3</u>

<u>Davis-Besse Unit 1</u>

<u>R. E. Ginna Unit 1</u>

<u>Oconee Unit 1</u>

Point Beach Unit 2

Surry Unit 1

<u>Surry Unit 2</u>

<u>Three Mile Island Unit 1</u>

<u>Turkey Point Unit 3</u>

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1	
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements	
Plant: Arkansas Nuclear One Unit 1	
Question I: Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II: Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach 1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2	
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.	
Question III: If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response: Not applicable (see Question I and II above)	
Question IV: If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response: Not applicable (see Question I and II above)	

NOTES: BAW-10006A, Revision 3: Surveillance Program Description (ASTM E 185-70)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Arkansas Nuclear One Unit 1					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
<u>LIMITING BELTLINE WELD</u> WF-112	70 (6)	8, approx.	48	43	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
<u>LIMITING BELTLINE PLATE OR FORGING</u> C5120-1	123 (7)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the ANO Unit 1 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) Result of fracture analysis presented in BAW-2075, Revision 1, demonstrate that the most limiting low upper-shelf welds have irradiated fracture toughness characteristics which will assure adequate margins of safety in accordance with the requirements of 10CFR50, Appendix G.

(6) BAW-1803

(7) BAW-1820

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Arkansas Nuclear One Unit 1								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
AYN 131	74,33,62 42,58,69	ND	ND	ND	ND	+3	Est. (2)	(1,3)
PLATE								
C5120-2	55,53,49	-16	+12	+5	-10	-10	NB-2331	(1,3,6)
C5114-2	40,50,36	+14	+42	+30	-10	-10	NB-2331	(1,3,6)
C5120-1	56,48,54	-6	+19	+20	-10	-10	NB-2331	(1,3,6)
C5114-1	57,40,57	+10	+40	+35	0	0	NB-2331	(1,3,6)
WELD								
WF-182-1	36,33,44	ND	ND	ND	ND	-5	Est. (4)	(1,5)
WF-112	35,40,30	ND	ND	ND	ND	-5	Est. (4)	(1,5)
SA-1788	40,38,36	ND	ND	ND	ND	-5	Est. (4)	(1,5)
WF-18	45,46,38	ND	ND	ND	ND	-5	Est. (4)	(1,5)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-1820

(2) BAW-10046P, pp 3-17, -18; mean of most conservative value for each of 24 cases.

(3) C_v(+10F) values are for 60 hr stress-relief; other values for 40 hr stress-relief.

(4) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(5) C_v(+10F) values are for 48 hr stress-relief.

(6) Mt. Vernon qualification test data.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: Arkansas Nuclear One Unit 1		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
AYN 131	1580±20F-5h/WQ; 1250±20F-14h/WQ; 1100-1150F-20:06h/FC (cumul.)	(1,2)
C5120-2	1550-1600F-4½h/BQ; 1200-1225F-5h/BQ; 1100-1150F-28½h/FC (cumul.)	
C5114-2	1550-1600F-4½h/BQ; 1200-1225F-5h/BQ; 1100-1150F-28½h/FC (cumul.)	
C5120-1	1550-1600F-4½h/BQ; 1200-1225F-5h/BQ; 1100-1150F-26¼h/FC (cumul.)	
C5114-1	1550-1600F-4½h/BQ; 1200-1225F-5h/BQ; 1100-1150F-26¼h/FC (cumul.)	
WF-182-1	1100-1150F-19h/FC (cumul.)	
WF-112	1100-1150F-24h/FC (cumul.)	
SA-1788	1100-1150F-20h/FC (cumul.)	
WF-18 (US Long.)	1100-1150F-28 h/FC (cumul.)	
WF-18 (LS Long.)	1100-1150F-26¼h/FC (cumul.)	
SURVEILLANCE MATERIALS		
C5114-1	1550-1600F-4½h/BQ; 1200-1225F-5h/BQ; 1100-1150F-29h/FC	(1)
C5114-2	1550-1600F-4½h/BQ; 1200-1225F-5h/BQ; 1100-1150F-29h/FC	
WF-193	1100-1150F-29h/FC	

NOTES:

(1) BAW-1820
(2) Additional stress relief information per Mt. Vernon process drawing.
(3) WQ - water quench
BQ - brine quench
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA					
Plant: Arkansas Nuclear One Unit 1					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Lower NB Forging	AYN 131, 528360	NB to US Circ.: WF-182-1	821T44	8754	(1)
US Plate	C5120-2	US to LS Circ.: WF-112	406L44	8688	
US Plate	C5114-2	LS to Dutch Circ.: SA-1788	61782	8754	
LS Plate	C5120-1	US Longit.: WF-18	8T1762	8650	
LS Plate	C5114-1	LS Longit.: WF-18	8T1762	8650	

NOTES: (1) BAW-1820
(2) NB - Nozzle Belt
US - Upper Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Arkansas Nuclear One Unit 1				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
C5114-1 C5114-2	WF-193	406L44	8773	(1)

NOTES: (1) BAW-1820

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)										
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>										
Plant: Arkansas Nuclear One Unit 1										
Column 1	Column 2									C. 3
Material	Chemical Composition Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
<u>BELTLINE MATERIAL</u>										
AYN 131	0.27	0.64	0.009	0.015	0.21	0.32	0.70	0.66	0.03	(1)
C5120-2	0.22	1.41	0.014	0.013	0.18	0.18	0.55	0.53	0.17	(1)
C5114-2	0.21	1.32	0.01[illegible]	0.016	0.20	0.19	0.52	0.57	0.15	(1)
C5120-1	0.22	1.41	0.014	0.013	0.18	0.18	0.55	0.53	0.17	(1)
C5114-1	0.21	1.32	0.010	0.016	0.20	0.19	0.52	0.57	0.15	(1)
WF-182-1	0.08	1.69	0.014	0.013	0.45	0.14	0.63	0.40	0.24	(2)
WF-112	0.08	1.47	0.016	0.015	0.54	0.07	0.59	0.40	0.31	(2)
WF-18	0.09	1.45	0.004	0.017	0.39	0.12	0.55	0.41	[illegible].20	(2)
<u>SURVEILLANCE MATERIALS</u>										
C5114-1	0.21	1.32	0.010	0.016	0.20	0.19	0.52	0.57	0.15	(3)
C5114-2	0.21	1.32	0.010	0.016	0.20	0.19	0.52	0.57	0.15	(3)
WF-193	0.09	1.49	0.016	0.016	0.51	0.06	0.59	0.39	0.28	(3)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) BAW-1820
(2) BAW-2121P
(3) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Arkansas Nuclear One Unit 1
Cold Leg Temperature (T_{cold}): 565 F (See Figure 4-1)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_VUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2: Not applicable
References: None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b

Subject:	Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant:	Arkansas N[illegible]ar One Unit 1

Were surveillance results used in determining C_vUSE? Yes ☐ No ✓

Were surveillance results used in determining RT_{NDT}? Yes ✓ No ☐

If any "yes" boxes were checked above, state how the surveillance results were used:

Determination of RT_{NDT} per Regulatory Guide 1.99, Revision 2, Position 2, for preparation of pressure-temperature limit curves for WF-182-1 and WF-112 weld materials only.

Reference: BAW-2075, Revision 1

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c							
Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects							
Plant: Arkansas Nuclear One Unit 1							
Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?							
Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?							
Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (3)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (5)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (5)
AYN 131	---	ND	ND	--	ND	ND	--
C5120-2	---	ND	ND	--	ND	ND	--
C5114-2	4.28E+18	0(1)	115	No	17(1)	22(1)	No
C5120-1	---	ND	ND	--	ND	ND	--
C5114-1	7.27E+17	10(2)	72	No	14(2)	19(2)	No
	1.03E+19	66(2)	140	No	11(2)	23(2)	No
	1.46E+19	38(2)	151	No	16(2)	25(2)	No
WF-182-1	1.96E+18	127(3)	151	No	6(3)	17(4)	No
	5.92E+18	125(3)	200	No	13(3)	22(4)	No
	1.29E+19	175(3)	237	No	8(3)	26(4)	No
	9.62E+18	150(4)	223	No	16(6)	24(4)	No
WF-112	1.50E+18	78(3)	157	No	9(3)	19(6)	No
	8.95E+18	191(3)	250	No	12(3)	27(6)	No
	9.86E+18	185(3)	256	No	12(3)	27(6)	No
	8.21E+18	204(3	246	No	29(3)	33(7)	No
SA-1788	---	ND	ND	--	ND	ND	--
WF-18	---	ND	ND	--	ND	ND	--

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-1698
(2) BAW-2075, Revision 1
(3) BAW-1803, Revision 1
(4) BAW-2125
(5) No statement required.
(6) BAW-2050
(7) BAW-1920P

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Crystal River Unit 3		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ✓ No ☐
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I an II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I an II above)	

NOTES: BAW-10100A: Surveillance Program Description (ASTM E 185-73)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Crystal River Unit 3					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
<u>LIMITING BELTLINE WELD</u> WF-70	70 (5)	5, approx.	48	43	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
<u>LIMITING BELTLINE PLATE OR FORGING</u> C4344-1	123 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Crystal River Unit 3 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-1820

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)

Subjec : 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements

Plant: Crystal River Unit 3

Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
AJZ 94	103,96,97 101,111,91	ND	ND	ND	ND	+3	Est. (2)	(1,3)
PLATE								
C4344-1	39,40,36	+50	+80	Not avail.	-10	+20	NB-2331	(1,3,7)
C4344-2	42,40,30	+30	+80	Not avail.	-10	+20	NB-2331	(1,3,7)
C4347-1	53,54,47	+20	+50	+45	-20	-10	NB-2331	(1,3,7)
C4347-2	43,53,63	+85	+105	+95	-20	+45	NB-2331	(1,3,7)
WELD								
SA-1769	36,35,38	ND	ND	ND	ND	-5	Est. (4)	(1,5)
WF-169-1	36,43,42 42,29,46	ND	ND	ND	ND	-5	Est. (4)	(1,5)
WF-8	45,38,30	ND	ND	ND	ND	-5	Est. (4)	(1,5)
WF-18	45,46,38	ND	ND	ND	ND	-5	Est. (4)	(1,5)
WF-70	39,35,44	ND	ND	ND	ND	+18	Eval.(6)	(1,5,8)
SA-1580	31,29,25 49,41,40	ND	ND	ND	ND	-5	Est. (4)	(1,5)
WF-154	41,37,43	ND	ND	ND	ND	-5	Est. (4)	(1,5)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-1820

(2) BAW-10046P, pp 3-17, -18; mean of most conservative value for each of 24 cases.

(3) C_v(+10F) values are for 60 hr stress-relief; other values for 27-40 hr stress-relief.

(4) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(5) C_v(+10F) values are for 48-80 hr stress-relief.

(6) BAW-2100

(7) Supplementary Mt. Vernon test of surveillance material.

(8) RT_{NDT} value for 40 hr stress-relief maximum.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Crystal River Unit 3

Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
AZJ 94	1590±20F-7h/WQ; 1270±20F-14h/WQ; 1100-1150F-22½h/FC (cumul.)	(1,2)
C4344-1	1550-1600F-4½h/BQ; 1175-1200F-6h/BQ; 1100-1150F-27h/FC (cumul.)	
C4344-2	1550-1600F-4½h/BQ; 1175-1200F-6h/BQ; 1100-1150F-27h/FC (cumul.)	
C4347-1	1550-1600F-4½h/BQ; 1250-1275F-5h/BQ; 1100-1150F-24¾h/FC (cumul.)	
C4347-2	1550-1600F-4½h/BQ; 1250-1275F-5h/BQ; 1100-1150F-24¾h/FC (cumul.)	
SA-1769	1100-1150F-19½h/FC (cumul.)	
WF-169-1	1100-1150F-19½h/FC (cumul.)	
WF-70	1100-1150F-20½h/FC (cumul.)	
WF-154	1100-1150F-27h/FC (cumul.)	
WF-8	1100-1150F-27h/FC (cumul.)	
WF-18	1100-1150F-27h/FC (cumul.)	
SA-1580	1100-1150F-24¾h/FC (cumul.)	
SURVEILLANCE MATERIALS		
C4344-1	1550-1600F-4½h/BQ; 1175-1200F-6h/BQ; 1100-1150F-27h/FC	(1)
C4344-2	1550-1600F-4½h/BQ; 1175-1200F-6h/BQ; 1100-1150F-27h/FC	
WF-209-1	1100-1150F-27h [cooling not reported]	
Atypical weld	1100-1150F-27h [cooling not reported]	

NOTES FOR TABLE 4 ARE ON FOLLOWING PAGE.

TABLE 4. (CONTINUED)

NOTES:

(1) BAW-1820
(2) Additional stress relief information per Mt. Vernon process drawing.
(3) WQ - water quench
BQ - brine quench
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Crystal River Unit 3

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Lower NB Forging	AZJ 94, 123V190	NB to US Circ.(ID 40%): SA-1769	71249	8738	(1)
US Plate	C4344-1	NB to US Circ.(OD 60%): WF-169-1	8T1554	8754	
US Plate	C4344-2	US to LS Circ.: WF-70	72105	8669	
LS Plate	C4347-1	LS to Dutch. Circ.: WF-154	406L44	8720	
LS Plate	C4347-2	US Longit.: WF-8	8T1762	8632	
		US Longit.: WF-18	8T1762	8650	
		LS Longit.: SA-1580	8T1762	8596	

NOTES: (1) BAW-1820
(2) NB - Nozzle Belt
US - Upper Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Crystal River Unit 3

Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
C4344-1 C4344-2	WF-209-1 Atypical	72105 Atypical	8773 8773	(1)

NOTES: (1) BAW-1820

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b. ¶ (5)										
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA										
Plant: Crystal River Unit 3										
Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
AZJ 94	0.26	0.65	0.007	0.016	0.24	0.34	0.72	0.62	ND	(1)
C4344-1	0.23	1.30	0.008	0.016	0.22	0.11	0.54	0.55	0.20	(1)
C4344-2	0.23	1.30	0.008	0.016	0.22	0.11	0.54	0.55	0.20	(1)
C4347-1	0.22	1.32	0.013	0.015	0.24	0.11	0.58	0.55	0.12	(1)
C4347-2	0.22	1.32	0.013	0.015	0.24	0.11	0.58	0.55	0.12	(1)
SA-1580	0.07	1.45	0.015	0.013	0.43	0.12	0.55	0.41	0.20	(2)
SA-1769	0.09	1.49	0.020	0.014	0.56	0.16	0.61	0.37	0.26	(2)
WF-8	0.06	1.45	0.009	0.009	0.53	0.12	0.55	0.41	0.20	(2)
WF-18	0.09	1.45	0.004	0.017	0.39	0.12	0.55	0.41	0.20	(2)
WF-70	0.09	1.63	0.018	0.009	0.54	0.10	0.59	0.40	0.35	(2)
WF-169-1	0.08	1.56	0.016	0.016	0.45	0.08	0.63	0.37	0.18	(2)
SURVEILLANCE MATERIALS										
C4344-1	0.23	1.30	0.008	0.016	0.22	0.11	0.54	0.55	0.20	(3)
C4344-2	0.23	1.30	0.008	0.016	0.22	0.11	0.54	0.55	0.20	(3)
Atypical	0.08	1.65	0.021	0.013	1.00	0.07	0.10	0.45	0.41	(3)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES FOR TABLE 7 ARE ON THE FOLLOWING PAGE.

TABLE 7. (CONTINUED)

NOTES:

(1) BAW-1820
(2) BAW-2121P
(3) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature

Plant: Crystal River Unit 3

Cold Leg Temperature (T_{cold}): 556 F (See Figure 4-1)

If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not applicable

References:

None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Crystal River Unit 3
Were surveillance results used in determining C_vUSE? Yes ☐ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No ☐
If any "yes" boxes were checked above, state how the surveillance results were used: Determination of RT_{NDT} using measured values for "atypical" weld material only. RT_{NDT} was also used for preparation of pressure-temperature limit curves.
Reference: BAW-2049

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Crystal River Unit 3

Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (1,3)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (5)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (5)
AZJ 94	---	ND	ND	--	ND	ND	--
C4344-1	1.17E+18	21(1)	98	No	5(1)	17(1)	No
	6.56E+18	126(1)	159	No	18(1)	25(1)	No
	7.50E+18	97(1)	164	No	22(1)	26(1)	No
	1.08E+19	128(1)	179	No	23(1)	28(1)	No
C4344-2	6.56E+18	127(2)	159	No	17(2)	25(2)	No
C4347-1	---	ND	ND	--	ND	ND	--
C4347-2	---	ND	ND	--	ND	ND	--
SA-1769	---	ND	ND	--	ND	ND	--
WF-169-1	---	ND	ND	--	ND	ND	--
WF-70	6.63E+18	135(3)	259	No	13(3)	25(4)	No
WF-154	---	ND	ND	--	ND	ND	--
WF-8	---	ND	ND	--	ND	ND	--
WF-18	---	ND	ND	--	ND	ND	--
SA-1580	---	ND	ND	--	ND	ND	--
Atypical	1.17E+18	28(1)	138	No	9(1)	25(1)	No
	6.56E+18	122(1)	216	No	16(1)	32(1)	No
	7.50E+18	119(1)	223	No	11(1)	32(1)	No
	1.08E+19	120(1)	242	No	15(1)	34(1)	No

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-2049
(2) BAW-1898
(3) BAW-1803, Revision 1
(4) BAW-1920p
(5) Statement not required

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION I	
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements	
Plant: Davis Besse Unit 1	
Question I: Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ✓ No □
Question II: Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No □
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.	
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.	
Question III: If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response: Not applicable (see Question I and II above)	
Question IV: If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response: Not applicable (see Question I and II above)	

NOTES: BAW-10100A: Surveillance Program Description
(ASTM E 185-73)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Davis-Besse Unit 1					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD					
WF-233	70 (2)	>32	NA	NA	NA
LIMITING BELTLINE PLATE OR FORGING					
BCC 241	118 (3)	>32	NA	NA	NA

NOTES: (1) Fluence values taken at ¼-thickness.

(2) BAW-1803

(3) BAW-1820

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Davis-Besse Unit 1								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad . RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
ADB 203	71,70,67 118,113,102	+48	+65	Not avail.	+50	+50	NB-2331	(1,3,6)
AKJ 233	ND	-15	+30	+15	+20	+20	NB-2331	(1,4,6)
BCC 241	ND	-14	+27	+5	+50	+50	NB-2331	(1,4,6)
WELD								
WF-232	25,31,35	ND	ND	ND	ND	-5	Est. (2)	(1,5)
WF-233	43,30,26	ND	ND	ND	ND	-5	Est. (2)	(1,5)
WF-182-1	36,33,44	+5	+62	Not avail.	-20	+2	NB-2331	(1,5)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-1820

(2) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(3) C_v(+10F) values are for 40 hr stress-relief; other values for unknown stress-relief.

(4) Values are for 15½ hr stress-relief.

(5) C_v(+10F) values are for 48 hr stress-relief.

(6) Supplementary Mt. Vernon tests of excess surveillance program material.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: Davis-Besse Unit 1		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
ADB 203	1590±10F-6h/WQ; 1240±10F-14h/WQ; 1100-1150F-15½h/FC (cumul.)	(1,2)
AKJ 233	1590±10F-4h/WQ; 1240±10F-6h/AC; 1100-1150F-15h/FC (cumul.)	
BCC 241	1590±10F-4h/WQ; 1240±10F-5h/AC; 1100-1150F-15h/FC (cumul.)	
WF-232	1100-1150F-14h/FC (cumul.)	
WF-233	1100-1150F-14h/FC (cumul.)	
WF-182-1	1100-1150F-15h/FC (cumul.)	
WF-232	1100-1150F-14½h/FC (cumul.)	
WF-233	1100-1150F-14½h/FC (cumul.)	
SURVEILLANCE MATERIALS		
BCC 241	1590±10F-4h/WQ; 1240±10F-5h/AC; 1100-1150F-15½h/FC	(1)
AKJ 233	1590±10F-4h/WQ; 1240±10F-6h/AC; 1100-1150F-15½h/FC	
WF-182-1	1100-1150F-15½h/FC	

NOTES:

(1) BAW-1820
(2) Additional stress relief information per MT. Vernon process drawing.
(3) WQ - water quench
AC - air cool
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA					
Plant: Davis-Besse Unit 1					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
NB Forging US Forging LS Forging	ADB 203, 123Y317 AKJ 233, 123X244 BCC 241, 5P4086	NB to US Circ.(ID 9%): WF-232 NB to US Circ.(OD 91%): WF-233 US to LS Circ.: WF-182-1 LS to Dutch Circ.(ID 12%): WF-232 LS to Dutch Circ.(OD 88%): WF-233	8T3914 T29744 821T44 8T3914 T29744	8790 8790 8754 8790 8790	(1)

NOTES: (1) BAW-1820
(2) NB - Nozzle Belt
US - Upper Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Davis-Besse Unit 1				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
BCC 241, 5P4086 AKJ 233, 123X244	WF-182-1	821T44	8754	(1)

NOTES: (1) BAW-1820

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Davis-Besse Unit 1

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
ADB 203	0.23	0.70	0.007	0.009	0.29	0.39	0.68	0.63	0.04	(1)
AKJ 233	0.26	0.68	0.004	0.006	0.30	0.38	0.77	0.64	0.04	(1)
BCC 241	0.22	0.63	0.011	0.011	0.27	0.32	0.81	0.63	0.02	(1)
WF-232	0.06	1.30	0.016	0.011	0.47	0.11	0.64	0.37	0.18	(2)
WF-233	0.05	1.45	0.021	0.015	0.42	0.08	0.68	0.44	0.29	(2)
WF-182-1	0.08	1.69	0.014	0.013	0.45	0.14	0.63	0.40	0.24	(2)
SURVEILLANCE MATERIALS										
BCC 241	0.22	0.63	0.011	0.011	0.27	0.32	0.81	0.63	0.02	(3)
AKJ 233	0.26	0.68	0.004	0.006	0.30	0.38	0.77	0.64	0.04	(3)
WF-182-1	0.09	1.69	0.014	0.013	0.41	0.15	0.63	0.40	0.21	(3)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) BAW-1820
(2) BAW-2121P
(3) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Davis-Besse Unit 1
Cold Leg Temperature (T_{cold}): 556 F (See Figure 4-2)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2: Not applicable
References: None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Davis-Besse Unit 1
Were surveillance results used in determining C_vUSE? Yes □ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No □
If any "yes" boxes were checked above, state how the surveillance results were used: Determination of RT_{NDT} per Regulatory Guide 1.99, Revision 2, Position 2, for preparation of pressure-temperature limit curves for WF-182-1 and WF-233 weld materials only.
Reference: BAW-2125

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Davis-Besse Unit 1

Question I. Does measured ΔRT_{NDT} exceed $\Delta RT_{NDT} + 2\sigma$ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_VUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (3)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (4)	Measured C_VUSE Drop	Predicted C_VUSE Drop	Question II If "yes" see Note (4)
ADB 203	---	ND	ND	--	ND	--	--
AKJ 233	1.29E+19	2(1)	56	No	13(1)	17(1)	No
BCC 241	1.96E+18	0(2)	24	--	9(2)	9(2)	No
	5.92E+18	0(2)	34	--	9(2)	11(2)	No
	1.29E+19	28(2)	44	No	4(2)	14(2)	No
	9.62E+18	3(2)	40	No	5(2)	12(2)	No
WF-232	---	ND	ND	--	ND	ND	--
WF-233	4.67E+18	191(3)	211	No	18(3)	23	No
	1.08E+19	187(3)	257	No	24(3)	28	No
	1.21E+19	222(3)	263	No	19(3)	28	No
WF-182-1	1.96E+18	127(3)	152	No	6(3)	17(2)	No
	5.92E+18	125(3)	200	No	13(3)	22(2)	No
	1.29E+19	175(3)	237	No	8(3)	26(2)	No
	9.62E+18	150(2)	223	No	16(2)	25(2)	No

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-1882, Revision 1
(2) BAW-2125
(3) BAW-1803, Revision 1
(4) No Statement required.

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: R. E. Ginna		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ✓(1) No □
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No □
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: (1) Robert E. Ginna Final Safety Analysis Report, Revision 6, Docket No. 50-244, December 1990.

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: R. E. Ginna					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD SA-847	70 (5)	4, approx.	41	37	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
LIMITING BELTLINE PLATE OR FORGING 125S255	124 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the R. E. Ginna analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements

Plant: R. E. Ginna

Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
<u>FORGING</u>								
123P118	30, average	-3	+9	+4	+30	+30	NB-2331	(1,4)
125S255	45,121,112 79,102,60, 97	-56	-48	-47	+20	+20	NB-2331	(1,2,4)
125P666	105,115,112 108,77,112	-23	-5	-1	+40	+40	NB-2331	(1,2,4)
<u>WELD</u>								
SA-1101	45,45,46	ND	+70	ND	-70	+10	NB-2331	(2,5,6)
SA-847	58,60,36	ND	ND	ND	ND	-5	Est. (3)	(2,5)
SA-848	54,56,59	ND	ND	ND	ND	-5	Est. (3)	(2,5)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) Supplier test report data.

(2) BAW-2150

(3) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(4) Values are for 30 hr stress relief.

(5) C_v(+10F) values are for 8 - 6 hr cycles stress relief.

(6) EPRI NP-373; C_v 50 ft-lb, Drop Weight, and RT_{NDT} values.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>

Plant: R. E. Ginna

Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
<u>BELTLINE MATERIALS</u>		
123P118VA1	1550F-11h/WQ; 1220F-22h; 1125F-11h (min)/FC	(1,2)
125S255VA1	1550F-15½h/WQ; 1210F-18h/AC; 1125F-10½h (min)/FC	
125P666VA1	1550F-9h/WQ; 1220F-12h/AC; 1125F-10½h (min)/FC	
SA-1101	1125F-9h (min)/FC	
SA-847	1125F-10½h (min)/FC	
SA-848	1125F-9¾h (min)/FC	
<u>SURVEILLANCE MATERIALS</u>		
125S255VA1	1550F-15½h/WQ; 1220F-18h/AC; 1100F-11½h/FC	(1)
125P666VA1	1550F-9h/WQ; 1220F-12h/AC; 1100F-11½h/FC	
SA-1036	1100F-11½h/FC	

NOTES:

(1) BAW-2150
(2) Additional stress relief information per Mt. Vernon fabrication process sheets
(3) WQ - water quench
AC - air cool
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>					
Plant: R. E. Ginna					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
NB Forging IS Forging LS Forging	123P118 125S255 125P666	NB to IS Circ.: SA-1101 IS to LS Circ.: SA-847 LS to Dutch Circ.: SA-848	71249 61782 61782	8445 8350 8373	(1,2)

NOTES: (1) BAW-2150
(2) Mt. Vernon fabrication process sheets
(3) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: R. E. Ginna				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
125S255 125P666	SA-1036	61782	8436	(1)

NOTES: (1) BAW-2150

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A: Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: R. E. Ginna

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
123P118VA1	0.20	0.64	0.010	0.008	0.23	0.41	0.68	0.60	ND	(1)
125S255VA1	0.18	0.66	0.010	0.006	0.23	0.34	0.68	0.58	0.07	(2,5)
125P666VA1	0.19	0.67	0.010	0.011	0.20	0.37	0.68	0.57	0.05	(2,5)
SA-1101	0.07	1.28	0.021	0.014	0.52	0.16	0.60	0.37	0.26	(3)
SA-847	0.08	1.34	0.012	0.012	0.45	0.08	0.54	0.38	0.25	(3)
SA-848	0.08	1.44	0.012	0.011	0.51	0.08	0.54	0.38	0.25	(2)
SURVEILLANCE MATERIALS										
125S255VA1	0.18	0.66	0.010	0.007	0.23	0.33	0.69	0.58	0.07	(4)
125P666VA1	0.19	0.67	0.010	0.011	0.20	0.37	0.69	0.57	0.05	(4)
SA-1036	0.08	1.41	0.012	0.016	0.59	0.09	0.56	0.36	0.23	(4)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) Supplier Material Test Report
(2) BAW-2150
(3) BAW-2121P
(4) BAW-1543, Revision 1
(5) Copper content based on surveillance material data.

TABLE 8. GENEPIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature

Plant: R. E. Ginna

Cold Leg Temperature (T_{cold}): 545 F (See Figure 4-3,

If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not applicable

References:

None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: R. E. Ginna
Were surveillance results used in determining C_vUSE? Yes □ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No □
If any "yes" boxes were checked above, state how the surveillance results were used: R. E. Ginna - Application for Amendment Docket 50-244.
References: Letter to A. R. Johnson from R. C. Mecredy dated February 15, 1991.

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: R. E. Ginna

Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (2)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (3)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (3)
123P118	---	ND	ND	--	ND	ND	--
125S255	6.53E+18	0(1)	73	No	0(1)	20	No
	1.02E+19	0(1)	78	No	0(1)	23	No
	1.78E+19	0(1)	85	No	0(1)	26	No
125P666	6.53E+18	25(1)	55	No	23(1)	23	No
	1.02E+19	25(1)	62	No	13(1)	26	No
	1.78E+19	30(1)	70	No	40(1)	29	Yes
SA-1101	7.01E+18	164(2)	195	No	4(2)	21	No
	1.23E+19	178(2)	220	No	18(2)	24	No
SA-847	---	ND	ND	--	ND	ND	--
SA-848	---	ND	ND	--	ND	ND	--

NOTES FOR TABLE 10 (IN PARENTHESIS ABOVE) ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) WCAP-10086
(2) BAW-1803, Revision 1
(3) The only instance where a measured "shift" or "drop" exceeds that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2, is for a "drop" for base metal. The requirements of 10CFR50, Appendix G, were not violated. The use of "drop" data is only to indicate if beltline material has fallen below 50 ft-lb. Since this has not occurred, the effect of these surveillance results are not significant.

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Oconee Unit 1		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: BAW-10006A, Revision 3: Surveillance Program Description (ASTM E 185-70)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a

Subject: 10CFR50, Appendix G, C_vUSE Requirements

Plant: Oconee Unit 1

Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
<u>LIMITING BELTLINE WELD</u> SA-1229	70 (5)	14, approx.	49	46	Analysis per 10CFR50, Appendix G, Section V.C.3. Is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
<u>LIMITING BELTLINE PLATE OR FORGING</u> C2197-2	91 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Oconee Unit 1 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements

Plant: Oconee Unit 1

Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
AHR 54	87,54,112 80,95,107	ND	ND	ND	ND	+3	Est. (2)	(1)
PLATE								
C2197-2	54,58,65 39,45,26	ND	ND	ND	≤+10	+1	Est. (3)	(1,4)
C3265-1	34,64,27 37,65,63	ND	ND	ND	≤+10	+1	Est. (3)	(1,5)
C3278-1	35,29,53 65,94,60	ND	ND	ND	≤+10	+1	Est. (3)	(1,5)
C2800-1	44,39,36 36,39,39	ND	ND	ND	≤+10	+1	Est. (3)	(1,4)
C2800-2	ND	ND	ND	ND	0	+1	Est. (3)	(1)

WELD								
SA-1135	56,44,55	ND	ND	ND	ND	-5	Est. (6)	(1,7)
SA-1229	55,45,40	ND	ND	ND	ND	-5	Est. (6)	(1,7)
WF-25	38,28,49	ND	ND	ND	ND	-5	Est. (6)	(1,7)
SA-1585	31,32,31 50,54,51	ND	ND	ND	ND	-5	Est. (6)	(1,8)
SA-1073	40,45,39	ND	ND	ND	ND	-5	Est. (6)	(1,9)
SA-1493	41,35,40	ND	ND	ND	ND	-5	Est. (6)	(1,7)
SA-1430	54,52,53	ND	ND	ND	ND	-5	Est. (6)	(1,7)
SA-1426	46,31,36 35,45,45	ND	ND	ND	ND	-5	Est. (6)	(1,10)

NOTES TO TABLE 3:

(1) BAW-1820
(2) BAW-10046P, pp 3-17, -18; mean of most conservative value for each of 24 cases.
(3) BAW-10046P, pp 3-18; mean of most conservative value for each of 13 cases.
(4) Values are for 60 hr stress-relief.
(5) Values are for 40 hr stress-relief.
(6) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.
(7) Values are for 48 hr stress-relief.
(8) C_v(+10F) test results from center and surface of test block; 80 hr stress-relief.
(9) C_v(+10F) values are for 8 6 hr cycles stress relief.
(10) C_v(+10F) test results from center and surface of test block; 48 hr stress-relief.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Oconee Unit 1

Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
AHR 54	1600±20F-5h/WQ; 1250±20F-15h/WQ; 1100-1150F-78¾h/FC (cumul.)	(1,2)
C2197-2	1600-1650F-9½h/BQ; 1200-1225F-9½h/BQ; 1100-1150F-46½h/FC (cumul.)	
C3265-1	1600-1650F-9¾h/BQ; 1200-1220F-9½h/BQ; 1100-1150F-50h/FC (cumul.)	
C3278-1	1600-1650F-9¾h/BQ; 1200-1225F-9½h/BQ; 1100-1150F-50h/FC (cumul.)	
C2800-1	1600-1650F-9½h/BQ; 1200-1225F-9½h/BQ; 1100-1150F-49h/FC (cumul.)	
C2800-2	1600-1650F-9½h/BQ; 1200-1225F-9½h/BQ; 1100-1150F-49h/FC (cumul.)	
SA-1135	1100-1150F-43½h/FC (cumul.)	
SA-1229	1100-1150F-43½h/FC (cumul.)	
WF-25	1100-1150F-43½h/FC (cumul.)	
SA-1585	1100-1150F-48h/FC (cumul.)	
WF-9	1100-1150F-43½h/FC (cumul.)	
SA-1073	1100-1150F-46½h/FC (cumul.)	
SA-1493	1100-1150F-50h/FC (cumul.)	
SA-1430	1100-1150F-49h/FC (cumul.)	
SA-1426	1100-1150F-49h/FC (cumul.)	
SURVEILLANCE MATERIALS		
C3265-1	1600-1650F-9¾h/BQ; 1200-1220F-9½h/BQ; 1100-1150F-40h/FC	(1)
C2800-2	1600-1650F-9½h/BQ; 1200-1225F-9½h/BQ; 1100-1150F-40h/FC	
WF-112	1100-1150F-40h/FC	

NOTES FOR TABLE 4 ARE ON FOLLOWING PAGE

TABLE 4 (CONTINUED)

NOTES:

(1) BAW-1820
(2) Additional stress relief information per Mt. Vernon process drawing.
(3) WQ - water quench
BQ - brine quench
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Oconee Unit 1

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Lower NB Forging	AHR 54, ZV 2861	NB to IS Circ.: SA-1135	61782	8457	(1)
IS Plate	C2197-2	IS to US Circ.(ID 61%): SA-1229	71249	8492	
US Plate	C3265-1	IS to US Circ.(OD 39%): WF-25	299L44	8650	
US Plate	C3278-1	US to LS Circ.: SA-1585	72445	8597	
LS Plate	C2800-1	LS to Dutch Circ.: WF-9	72445	8632	
LS Plate	C2800-2	IS Longit.: SA-1073	1P0962	8445	
		US Longit.: SA-1493	8T1762	8578	
		LS Longit.: SA-1430	8T1762	8553	
		LS Longit.: SA-1426	8T1762	8553	

NOTES: (1) BAW-1820
(2) NB - Nozzle Belt
IS - Intermediate Shell
US - Upper Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Oconee Unit 1				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
C3265-1 C2800-2	WF-112	406L44	8688	(1)

NOTES: (1) BAW-1820

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE E[illegible]LIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>

Plant: Oconee Unit 1

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
<u>BELTLINE MATERIALS</u>										
AHR 54	0.18	0.64	0.006	0.010	0.29	0.31	0.65	0.57	0.16	(1)
C2197-2	0.21	1.28	0.008	0.010	0.17	---	0.50	0.46	0.15	(1)
C3265-1	0.21	1.42	0.015	0.015	0.23	0.17	0.50	0.49	0.10	(1)
C3278-1	0.19	1.26	0.010	0.016	0.23	0.11	0.60	0.47	0.12	(1)
C2800-1	0.20	1.40	0.012	0.017	0.20	0.13	0.63	0.50	0.11	(1)
C2800-2	0.20	1.40	0.012	0.017	0.20	0.13	0.63	0.50	0.11	(1)
SA-1073	0.10	1.38	0.025	0.017	0.51	0.11	0.64	0.43	0.21	(2)
SA-1135	0.08	1.45	0.011	0.013	0.49	0.08	0.54	0.38	0.25	[illegible]
SA-1229	0.06	1.56	0.021	0.012	0.43	0.16	0.61	0.37	0.26	(2)
SA-1426	0.08	1.53	0.017	0.013	0.43	0.12	0.55	0.41	0.20	(2)
SA-1430	0.08	1.43	0.017	0.015	0.43	0.12	0.55	0.41	0.20	(2)
SA-1493	0.08	1.51	0.017	0.010	0.46	0.12	0.55	0.41	0.20	(2)
SA-1585	0.08	1.45	0.016	0.016	0.51	0.09	0.59	0.38	0.21	(2)
WF-9	0.08	1.45	0.016	0.016	0.51	0.09	0.59	0.38	0.21	(1)
WF-25	0.09	1.60	0.015	0.016	0.50	0.09	0.68	0.42	0.35	(2)
<u>SURVEILLANCE MATERIALS</u>										
C3265-1	0.21	1.42	0.015	0.015	0.23	0.17	0.50	0.49	0.10	(3)
C2800-2	0.20	1.40	0.012	0.017	0.20	0.13	0.63	0.50	0.11	(3)
WF-112	0.08	1.47	0.016	0.015	0.54	0.07	0.59	0.40	0.32	(3)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

TABLE 7. (CONTINUED)

NOTES:

(1) BAW-1820
(2) BAW-2121P
(3) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Oconee Unit 1
Cold Leg Temperature (T_{cold}): 556 F (See Figure 4-1)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2: Not applicable
References: None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Oconee Unit 1
Were surveillance results used in determining C_vUSE? Yes ☐ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No ☐
If any "yes" boxes were checked above, state how the surveillance results were used: Determination of RT_{NDT} per Regulatory Guide 1.99, Revision 2, Position 2, for prepartation of pressure-temperature limit curves for WF-25 weld material only.
Reference: BAW-2050

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Oconee Unit 1

Question I. Does measured ΔRT_{NDT} exceed $\Delta RT_{NDT} + 2\sigma$ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (3)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (4)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (4)
AHR 54	---	ND	ND	--	ND	ND	--
C2197-2	---	ND	ND	--	ND	ND	--
C3265-1	1.50E+18	15(1)	65	No	1(1)	13(1)	No
	8.95E+18	34(1)	97	No	2(1)	20(1)	No
	9.86E+18	39(1)	99	No	12(1)	21(1)	No
C3278-1	---	ND	ND	--	ND	ND	--
C2800-1	---	ND	ND	--	ND	ND	--
C2800-2	8.30E+17	18(2)	57	No	0(2)	13(2)	No
SA-1135	1.03E+19	142(3)	240	No	21(3)	31(5)	No
SA-1229	---	ND	ND	--	ND	ND	--
WF-25	1.07E+18	124(6)	148	No	17(3)	24(6)	No
	8.66E+18	203(6)	261	No	31(3)	34(6)	No
	7.79E+18	214(3)	263	No	25(3)	30(7)	No
SA-1585	5.10E+18	148(3)	188	No	22(3)	24(7)	No
WF-9	---	ND	ND	--	ND	ND	--
SA-1073	---	ND	ND	--	ND	ND	--
SA-1493	---	ND	ND	--	ND	ND	--
SA-1430	---	ND	ND	--	ND	ND	--
SA-1426	---	ND	ND	--	ND	ND	--

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-2050
(2) BAW-1421, Revision 1
(3) BAW-1803, Revision 1
(4) Statement not required.
(5) BAW-1920P
(6) BAW-1901
(7) BAW-1910P

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Oconee Unit 2		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: BAW-10006A, Revision 3: Surveillance Program Description (ASTM E 185-70)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Oconee Unit 2					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD WF-25	70 (5)	4, approx.	46	43	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
LIMITING BELTLINE PLATE OR FORGING AMX 77	124 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Oconee Unit 2 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements

Plant: Oconee Unit 2

Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10°F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
<u>FORGING</u>								
AMX 77	90,121,106 103,91,128	ND	ND	ND	ND	+3	Est. (2)	(1,4)
AAW 163	ND	-40	-10	-15	+20	+20	NB-2331	(1,5,7)
AWG 164	ND	-75	-45	-50	+20	+20	NB-2331	(1,5,7)
<u>WELD</u>								
WF-154	41,37,43	ND	ND	ND	[illegible]	-5	Est. (3)	(1,6)
WF-25	38,28,49	ND	ND	ND	[illegible]	-5	Est. (3)	(1,6)
WF-112	35,40,30	ND	ND	ND	ND	-5	Est. (3)	(1,6)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-1820
(2) BAW-10046P, pp 3-17, -18; mean of most conservative value for each of 24 cases.
(3) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.
(4) Values are for 60 hr stress-relief.
(5) Values are for 40 hr stress-relief.
(6) C_v(+10F) values are for 48 hr stress-relief.
(7) Supplier test report data.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>		
Plant: Oconee Unit 2		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
<u>BELTLINE MATERIALS</u>		
AMX 77	1580±20F-7h/WQ; 1240±20F-14h/WQ; 1100-1150F-53½h/FC (cumul.)	(1,2)
AAW 163	1590±20F-4h/WQ; 1260±20F10h/WQ; 1100-1150F-41h/FC (cumul.)	
AWG 164	1590±10F-4h/WQ; 1260±20F-10h/WQ; 1100-1150F-41h/FC (cumul.)	
WF-154	1100-1150F-32¾h/FC (cumul.)	
WF-25	1100-1150F-41h/FC (cumul.)	
WF-112	1100-1150F-39¾h/FC (cumul.)	
<u>SURVEILLANCE MATERIALS</u>		
AAW 163	1590±20F-4h/WQ; 1260±20F-10h/WQ; 1100-1150F-33h/FC	(1)
AWG 164	1590±20F-4h/WQ; 1260±20F-10h/WQ; 1100-1150F-33h/FC	
WF-209-1	1100-1150F-33h/FC	

NOTES:

(1) BAW-1820
(2) Additional stress relief information per Mt. Vernon process drawing.
(3) WQ - water quench
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>

Plant: Oconee Unit 2

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Lower NB Forging	AMX 77, 123T382	NB to US Circ.: WF-154	406L44	8720	(1)
US Forging	AAW 163, 3P2359	US to LS Circ.: WF-25	299L44	8650	
LS Forging	AWG 164, 4P1885	LS to Dutch Circ.: WF-112	406L44	8688	

NOTES: (1) BAW-1820
(2) NB - Nozzle Belt
US - Upper Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Oconee Unit 2				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
AAW 163, 3P2359 AWG 164, 4P1885	WF-209-1	72105	8773	(1)

NOTES: (1) BAW-1820

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)										
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA										
Plant: Oconee Unit 2										
Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
AMX 77	0.25	0.65	0.006	0.009	0.23	0.36	0.76	0.64	0.06	(1)
AAW 163	0.24	0.63	0.006	0.012	0.25	0.36	0.75	0.62	0.04	(1)
AWG 164	0.21	0.62	0.010	0.010	0.23	0.39	0.80	0.58	0.02	(1)
WF-25	0.09	1.60	0.015	0.016	0.50	0.09	0.68	0.42	0.35	(2)
WF-112	0.08	1.47	0.016	0.015	0.54	0.07	0.59	0.40	0.31	(2)
WF-154	0.07	1.54	0.013	0.016	0.42	0.07	0.59	0.40	0.31	(2)
SURVEILLANCE MATERIALS										
AAW 163	0.24	0.63	0.006	0.012	0.25	0.36	0.75	0.62	0.04	(3)
AWG 164	0.21	0.62	0.010	0.010	0.23	0.39	0.80	0.58	0.02	(3)
WF-209-1	0.11	1.55	0.022	0.010	0.65	0.09	0.58	0.39	0.36	(3)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) BAW-1820
(2) BAW-2121P
(3) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature

Plant: Oconee Unit 2

Cold Leg Temperature (T_{cold}): 556 F (See Figure 4-1)

If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not applicable

References:

None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b

Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Oconee Unit 2
Were surveillance results used in determining C_vUSE? Yes ☐ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No ☐
If any "yes" boxes were checked above, state how the surveillance results were used: Determination of RT_{NDT} per Regulatory Guide 1.99, Revision 2, Position 2, for prepartation of pressure-temperature limit curves for WF-25 and WF-154 weld materials only.
Reference: BAW-2051

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Oconee Unit 2

Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (2)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (3)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (3)
AMX 77	---	ND	ND	--	ND	ND	--
AAW 163	1.02E+18	0(1)	22	No	12(1)	11(1)	Yes
	3.37E+18	0(1)	36	No	--	15(1)	--
	1.21E+19	0(1)	55	No	19(1)	20(1)	No
AWG 164	---	ND	ND	--	ND	ND	--
WF-154	---	ND	ND	--	ND	ND	--
WF-25	1.07E+18	124(2)	148	No	17(2)	24(4)	No
	8.66E+18	203(2)	261	No	31(2)	34(4)	No
	7.79E+18	214(2)	263	No	25(2)	30(4)	No
WF-112	1.50E+18	78(2)	157	No	9(2)	19(5)	No
	8.95E+18	191(2)	250	No	12(2)	27(5)	No
	9.86E+18	185(2)	256	No	12(2)	27(5)	No
	8.21E+18	204(2)	246	No	29(2)	33(6)	No

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-2051

(2) BAW-1803, Revision 1

(3) The only instance where a measured "shift" or "drop" exceeds that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2, is for a "drop" for base metal and that was by 1 ft-lb and is not considered to be significant. At a higher fluence, the predicted "drop" for that material did not exceed the measured value. The requirements of 10CFR50, Appendix G, were not violated, and there being no further application of the "drop" data, the effect of these surveillance results are therefore not significant.

(4) BAW-1901

(5) BAW-2050

(6) BAW-1920P

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Oconee Unit 3		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: BAW-10006A, Revision 3: Surveillance Program Description (ASTM E 185-70)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Oconee Unit 3					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD WF-67	70 (5)	17, approx.	50	47	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
LIMITING BELTLINE PLATE OR FORGING AWS 192	90 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Oconee Unit 3 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-1820

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements

Plant: Oconee Unit 3

Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
4680	117,111,109 113,49,101	ND	ND	ND	ND	+3	Est. (2)	(1,4)
AWS 192	ND	-55	-30	-40	+40	+40	NB-2331	(1,5,8)
ANK 191	ND	-2	+20	-3	+40	+40	NB-2331	(1,5,8)
WELD								
WF-200	36,25,26	ND	ND	ND	ND	-5	Est. (3)	(1,6)
WF-67	29,35,30	ND	ND	ND	ND	-5	Est. (3)	(1,6)
WF-70	39,35,44	ND	ND	ND	ND	+18	Eval.(7)	(1,6,9)
WF-169-1	42,29,46	ND	ND	ND	ND	-5	Est. (3)	(1,6)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-1820

(2) BAW-10046P, pp 3-17, -18; mean of most conservative value for each of 24 cases.

(3) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(4) C_v(+10F) values are for 60 hr stress-relief.

(5) Values are for 25 hr stress-relief.

(6) C_v(+10F) values are for 48 hr stress-relief.

(7) BAW-2100

(8) Supplier test report data.

(9) RT_{NDT} value are for 40 hr stress-relief maximum.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: Oconee Unit 3		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
4680	1675F-7h/WQ; 1220F-15h/AC; 1100-1150F-26½h/FC (cumul.)	(1,2)
AWS 192	1590±20F-4h/WQ; 1240±20F-10h/WQ; 1100-1150F-29¾h/FC (cumul.)	
ANK 191	1590±20F-4h/WQ; 1250±20F-10h/WQ; 1100-1150F-29¾h/FC (cumul.)	
WF-200	1100-1150F-25h/FC (cumul.)	
WF-67	1100-1150F-29¾h/FC (cumul.)	
WF-70	1100-1150F-29¾h/FC (cumul.)	
WF-169-1	1100-1150F-28½h/FC (cumul.)	
SURVEILLANCE MATERIALS		
ANK 191	1590±20F-4h/WQ; 1250±20F-10h/WQ; 1100-1150F-30h/FC	(1)
AWS 192	1590±20F-4h/WQ; 1240±20F-10h/WQ; 1100-1150F-30h/FC	
WF-209-1	1100-1150F-30h/FC	

NOTES:

(1) BAW-1820
(2) Additional stress relief information per Mt. Vernon process drawing.
(3) WQ - water quench
AC - air cool
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>					
Plant: Oconee Unit 3					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Lower NB Forging US Forging LS Forging	4680 AWS 192, 522314 ANK 191, 522194	NB to US Circ.: WF-200 US to LS Circ.(ID 75%): WF-67 US to LS Circ.(OD 25%): WF-70 LS to Dutch Circ.: WF-169-1	821T44 72442 72105 8T1554	8773 8669 8669 8754	(1)

NOTES: (1) BAW-1820
(2) NB - Nozzle Belt
US - Upper Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Oconee Unit 3				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
ANK 191, 522194 AWS 192, 522314	WF-209-1	72105	8773	(1)

NOTES: (1) BAW-1820

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)										
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA										
Plant: Oconee Unit 3										
Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
4680	0.21	0.67	0.009	0.012	0.22	0.36	0.91	0.56	---	(1)
AWS 192	0.21	0.58	0.011	0.015	0.24	0.30	0.73	0.60	0.01	(1)
ANK 191	0.24	0.72	0.014	0.012	0.21	0.34	0.76	0.62	0.02	(1)
WF-67	0.08	1.55	0.021	0.016	0.58	0.09	0.60	0.39	0.24	(2)
WF-70	0.09	1.63	0.018	0.009	0.54	0.10	0.59	0.40	0.35	(2)
WF-169-1	0.08	1.56	0.016	0.016	0.45	0.08	0.63	0.37	0.18	(2)
WF-200	0.07	1.60	0.010	0.015	0.48	0.14	0.63	0.40	0.24	(2)
SURVEILLANCE MATERIALS										
ANK 191	0.24	0.72	0.014	0.012	0.21	0.34	0.76	0.62	0.02	(3)
AWS 192	0.21	0.58	0.011	0.015	0.24	0.30	0.73	0.60	0.01	(3)
WF-209-1	0.08	1.63	0.017	0.012	0.61	0.10	0.58	0.39	0.30	(3)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) BAW-1820
(2) BAW-2121P
(3) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Oconee Unit 3
Cold Leg Temperature (T_{cold}): 556 F (See Figure 4-1)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2: Not applicable
References: None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Oconee Unit 3
Were surveillance results used in determining C_vUSE? Yes ☐ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ☐ No ✓
If any "yes" boxes were checked above, state how the surveillance results were used:
Reference: None

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c							
Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects							
Plant: Oconee Unit 3							
Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?							
Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?							
Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (2)	Measured ΔRT_{NDT}	Predicted ΔRT_{NDT}+2σ	Question I If "yes" see Note (3)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (3)
4680	---	ND	ND	--	ND	ND	--
AWS 192	8.10E+17	63(1)	16	Yes	12(1)	4(1)	Yes
	3.12E+18	19(1)	28	No	13(1)	9(1)	Yes
	1.45E+19	45(1)	44	Yes	19(1)	12(1)	Yes
ANK 191	8.10E+17	9(1)	16	No	14(1)	8(1)	Yes
	3.12E+18	32(1)	28	Yes	21(1)	13(1)	Yes
	1.45E+19	31(1)	44	No	19(1)	17(1)	Yes
WF-200	---	ND	ND	--	ND	ND	--
WF-67	6.09E+18	160(2)	200	No	15(2)	23(4)	No
WF-70	6.63E+18	135(2)	259	No	13(2)	22(5)	No
WF-169-1	---	ND	ND	--	ND	ND	--

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-2128

(2) BAW-1803, Revision 1

(3) (a) The only instances where a measured "drop" exceeds that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2, is for base metal. The requirements of 10CFR50, Appendix G, were not violated.

(b) The only instances where a measured "shift" exceeds that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2, is for base metal.

For forging AWS 192, the first such finding was for material irradiated to 8.1×10^{17} nvt. This was not observed for material irradiated to 3.1×10^{18} nvt. Since the measured shift of the material irradiated to a higher neutron fluence did not exceed the predicted value, it is safe to conclude that the conservativeness of the Regulatory Guide method was not compromised. The second such finding, for material irradiated to 1.4×10^{19}, showed a difference of one degree F and is not taken as significant.

For forging ANK 191, there was one finding, for material irradiated to 3.1×10^{18} nvt. This was not observed for material irradiated to 1.4×10^{19} nvt. Since the measured shift of the material irradiated to a higher neutron fluence did not exceed the predicted value, it is safe to conclude that the conservativeness of the Regulatory Guide method was not compromised.

(c) As noted above, the "drop" data did not violate regulatory requirements, and there is no further application of the "drop" data. In the instances where the measured "shift" values exceeded the predicted values, it was shown above that the conservativeness of the Regulatory Guide was not compromised. It is concluded, therefore, that the effect of these surveillance results is not significant.

(4) BAW-1910P

(5) BAW-1920P

TABLE 1. GENERIC LETTER 92-0 : SE N 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requir		
Plant: Point Beach Unit 1		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or	Yes ☐ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: WCAP-7513: Surveillance Program Description
(ASTM E 185-66)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Point Beach Unit 1					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
<u>LIMITING BELTLINE WELD</u> SA-1101	70 (5)	5, approx.	42	39	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
<u>LIMITING BELTLINE PLATE OR FORGING</u> A9811-1	91 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Point Beach Unit 1 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Point Beach Unit 1								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
122P237	63,66,67 62,95,42	-22	+4	+15	+50	+50	NB-2331	(1,5)
PLATE								
A9811-1	50,41,51	-4	+24	ND	-30	+1	Est. (3)	(1,2,6)
C1423-1	103,51,88	-38	-16	ND	-20	+1	Est. (3)	(1,2,6)
WELD								
SA-1426	35,45,45 46,31,36	ND	ND	ND	ND	-5	Est. (4)	(2,7,9)
SA-1101	45,45,46	ND	+70	ND	-70	+10	NB-2331	(2,8)
SA-812	43,40,36	ND	ND	ND	ND	-5	Est. (4)	(2,7)
SA-775	48,45,44	ND	ND	ND	ND	-5	Est. (4)	(2,7)
SA-847	58,60,36	ND	ND	ND	ND	-5	Est. (4)	(2,7)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) Supplier test report data.
(2) BAW-2150
(3) BAW-10046P, pp 3-18; mean of most conservative value for each of 13 cases.
(4) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.
(5) Values are for 30 hr stress-relief.
(6) Values are for 50 hr stress-relief.
(7) C_v(+10F) values are for 8 - 6 hr cycles stress relief.
(8) EPRI NP-373; C_v 50 ft-lb, Drop Weight, and RT_{NDT} values.
(9) C_v(+10F) test results from center and surface of test block.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Point Beach Unit 1

Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
122P237VA1	1550F-11h/WQ; 1220F-22h/FC; 1125F-10½h/FC	(1,2,3)
A9811-1	1625-1675F-1h/in (min)/WQ; 1200-1250F-1h/in (min)/AC; 1100-1150F-10¾h (min)/FC	
C1423-1	1625-1675F-1h/in (min)/WQ; 1200-1250F-1h/in (min)/AC; 1100-1150F-10½h (min)/FC	
SA-1426	1125F-9h (min)/FC	
SA-1101	1125F-10¼ (min)/FC	
SA-812/SA-775	1125F-10¾h (min)/FC	
SA-847	1125F-10½h (min)/FC	
SURVEILLANCE MATERIALS		
A9811-1	1650F-7h/WQ; 1225F-7h/AC; 1125F-11¼h/FC	(1)
C1423-1	1650F-7h/WQ; 1225F-7h/AC; 1125F-10½h/FC	
SA-1263	1125F-11¼h/FC	

NOTES FOR TABLE 4 ARE ON FOLLOWING PAGE.

TABLE 4. (CONTINUED)

NOTES:

(1) BAW-2150
(2) Supplier Material Test Report
(3) Additional stress relief information per Mt. Vernon fabrication process sheets.
(4) WQ - water quench
AC - air cool
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>					
Plant: Point Beach Unit 1					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
NB Forging IS Plate LS Plate	122P237 A9811-1 C1423-1	NB to IS Circ.: SA-1426 IS to LS Circ.: SA-1101 IS Longit.(ID 27%): SA-812 IS Longit.(OD 73%): SA-775 LS Longit.: SA-847	8T1762 71249 1P0815 1P0661 61782	8553 8445 8350 8304 8350	(1,2)

NOTES: (1) BAW-2150
(2) Mt. Vernon fabrication process sheets
(3) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Point Beach Unit 1				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
A9811-1 C1423-1	SA-1263	72445	8504	(1)

NOTES: (1) BAW-2150

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)										
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA										
Plant: Point Beach Unit 1										
Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
122P237VA1	0.21	0.65	0.010	0.008	0.22	0.33	0.82	0.62	0.15(7)	(1)
A9811-1	0.20	1.42	0.010	0.020	0.25	ND	0.056(6)	0.49	0.20	(2,5)
C1423-1	0.20	1.36	0.016	0.020	0.25	ND	0.065(6)	0.46	0.12	(2,5)
SA-1426	0.08	1.53	0.017	0.013	0.43	0.12	0.55	0.41	0.20	(3)
SA-812	0.08	1.54	0.017	0.015	0.40	0.07	0.52	0.38	0.17	(3)
SA-775	0.08	1.52	0.024	0.019	0.46	0.06	0.63	0.45	0.19	(3)
SA-1101	0.07	1.28	0.021	0.014	0.52	0.16	0.60	0.37	0.26	(3)
SA-847	0.08	1.34	0.012	0.012	0.45	0.08	0.54	0.38	0.25	(3)
SURVEILLANCE MATERIALS										
A9811-1	0.19	1.42	0.010	0.020	0.25	ND	0.056(6)	0.48	0.20	(4)
C1423-1	0.21	1.37	0.014	0.019	0.25	ND	0.065(6)	0.46	0.12	(4)
SA-1263	0.09	1.47	0.019	0.024	0.49	0.13	0.57	0.39	0.22	(4)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES FOR TABLE 7 ARE ON THE FOLLOWING PAGE.

TABLE 7. (CONTINUED)

NOTES:

(1) Supplier Material Test Report
(2) BAW-2150
(3) BAW-2121P
(4) BAW-1543, Revision 3
(5) Copper and nickel contents based on surveillance material data.
(6) These values are suspect; verification of this information is planned.
(7) Estimated value based on review of similar materials.

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Point Beach Unit 1
Cold Leg Temperature (T_{cold}): 542 F (See Figure 4-4)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2: During the time span from approximately December 1, 1979 to October 1, 1983, the plant operated at approximately 80% power and a reduced system average temperature. This operation produced a cold leg temperature of approximately 511 F. This temperature was not considered in determination of embrittlement effects since the surveillance capsule results spanning this interval did not exhibit any significant variation from the expected values. See Section 4.
References:

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Point Beach Unit 1
Were surveillance results used in determining C_vUSE? Yes □ No ✓
Were surveillance results used in determining RT_{NDT}? Yes □ No ✓
If any "yes" boxes were checked above, state how the surveillance results were used:
References: None

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Point Beach Unit 1

Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (2)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (3)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (3)
122P237	---	ND	ND	--	ND	ND	--
A9811-1	6.20E+18	90(1)	110	No	18(1)	28	No
	7.58E+18	90(1)	115	No	15(1)	29	No
	2.10E+19	105(1)	140	No	7(1)	37	No
	2.11E+19	100(1)	140	No	12(1)	37	No
C1423-1	6.20E+18	50(1)	82	No	0(1)	22	No
	7.58E+18	50(1)	85	No	0(1)	23	No
	2.10E+19	50(1)	100	No	0(1)	30	No
	2.11E+19	50(1)	101	No	0(1)	30	No
SA-1426	---	ND	ND	--	ND	ND	--
SA-1101	7.01E+18	164(2)	195	No	4(2)	21	No
	1.23E+19	178(2)	220	No	18(2)	24	No
SA-812	---	ND	ND	--	ND	ND	--
SA-775	---	ND	ND	--	ND	ND	--
SA-847	---	ND	ND	--	ND	ND	--

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) WCAP-10736

(2) BAW-1803, Revision 1

(3) Statement not required.

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1

Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Point Beach Unit 2		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is not to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: WCAP-7712: Surveillance Program Description (ASTM E 185-66)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a						
Subject: 10CFR50, Appendix G, C_vUSE Requirements						
Plant: Point Beach Unit 2						
Column 1		Column 2	Column 3		Column 4	
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1	
			Column 3A	Column 3B		
			12/16/91	EOL		
LIMITING BELTLINE WELD SA-1484	70 (5)	5, approx.	43	40	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.	
LIMITING BELTLINE PLATE OR FORGING 123V500	124 (6)	>32	NA	NA	NA	

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Point Beach Unit 2 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Point Beach Unit 2								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropwt. Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
123V352	75,58,61 60,55,62	-27	+15	+25	+40	+40	NB-2331	(2,4)
123V500	111,80.81 111,108,118	-50	-25	-57	+40	+40	NB-2331	(1,2,4)
123W195	38,43,46 66,54,75,78	-30	-[illegible]	-10	+40	+40	NB-2331	(1,2,4)
WELD								
CE Weld SA-1484	Not avail. 40,52,41	Not avail ND	Not avail ND	Not avail ND	Not avail ND	-56 -5	10CFR50.61 Est. (3)	 (1)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-2150

(2) Supplier Test Reports

(3) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(4) Values are for 30 hr stress relief.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN SME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: Point Beach Unit 2		
Column 1	Column 2	Col. 3
Material	Heat Treatment ·	Notes
BELTLINE MATERIALS		
123V352VA1	1550F-11½h/WQ; 1220F-12h/FC; 1125F-½h (min)/FC	(1,2)
123V500VA1	1550F-9½h/WQ; 1200F-12h/AC; 1125F-¾h (min)/FC	
122W195VA1	1550F-8h/WQ; 1200F-12h/AC; 1125F-¾h (min)/FC	
CE Weld	Not available	
SA-1484	1125F-¾h (min)/FC	
SURVEILLANCE MATERIALS		
123V500VA1	1550F-9½h/WQ; 1200F-12h/AC; 1125F-12h/FC	(1)
122W195VA1	1550F-8h/WQ; 1200F-12h/AC; 1125F-12h/FC	
WF-193	1125F-11½h/FC	

NOTES:

(1) BAW-2150
(2) Additional stress relief information per Mt. Vernon fabrication process sheets.
(3) WQ - water quench
AC - air cool
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA					
Plant: Point Beach Unit 2					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
NB Forging IS Forging LS Forging	123V352 123V500 122W195	NB to IS Circ.: CE Weld IS to LS Circ.: SA-1484	Not avail. 72442	Not avail. 8579	(1,2)

NOTES: (1) BAW-2150
(2) Mt. Vernon fabrication process sheets
(3) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Point Beach Unit 2				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
123V500 122W195	WF-193	406L44	8773	(1)

NOTES: (1) BAW-2150

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Point Beach Unit 2

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
123V352VA1	0.20	0.68	0.010	0.010	0.24	0.34	0.73	0.59	0.15(3)	(1)
123V500VA1	0.18	0.66	0.010	0.008	0.24	0.34	0.70	0.57	0.09	(4,7)
122W195VA1	0.24	0.58	0.010	0.008	0.22	0.33	0.72	0.57	0.05	(4,7)
NB to IS	NA	NA	NA	NA	NA	NA	0.90	NA	0.27	(2,4)
SA-1484	0.08	1.52	0.018	0.015	0.42	0.09	0.60	0.39	0.24	(5)
SURVEILLANCE MATERIALS										
123V500VA1	0.20	0.65	0.009	0.009	0.24	0.35	0.71	0.59	0.09	(6)
122W195VA1	0.22	0.59	0.010	0.008	0.23	0.33	0.70	0.60	0.05	(6)
WF-193	0.08	1.40	0.014	0.013	0.55	0.07	0.59	0.39	0.25	(6)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES FOR TABLE 7 ARE ON THE FOLLOWING PAGE.

TABLE 7. (CONTINUED)

NOTES:

(1) Supplier Material Test Report

(2) Nozzle belt-to-intermediate shell weld was fabricated by Combustion Enginnering; information is not available (NA).

(3) Estimated value based on review of similar materials.

(4) BAW-2150

(5) BAW-2121P

(6) BAW-1543, Revision 3

(7) Copper content based on surveillance material data.

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Point Beach Unit 2
Cold Leg Temperature (T_{cold}): 542 F (See Figure 4-4)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.59, Revision 2: Not applicable
References: None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b

Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results

Plant: Point Beach Unit 2

Were surveillance results used in determining C_vUSE? Yes ☐ No ✓

Were surveillance results used in determining RT_{NDT}? Yes ☐ No ✓

If any "yes" boxes were checked above, state how the surveillance results were used:

References: None

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c							
Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects							
Plant: Point Beach Unit 2							
Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?							
Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?							
Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (3)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (2)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (2)
123V352	---	ND	ND	--	ND	ND	--
123V500	6.14E+18	30(1)	84	No	0	29(1)	No
	8.36E+18	30(1)	89	No	0	31(1)	No
	2.15E+19	70(1)	104	No	0	38(1)	No
	3.47E+19	76(1)	111	No	17	41(1)	No
122W195	6.14E+18	10(1)	54	No	10	17(1)	No
	8.36E+18	17(1)	59	No	0	19(1)	No
	2.15E+19	35(1)	71	No	5	23(1)	No
	3.47E+19	47(1)	75	No	11	26(1)	No
CE Weld	---	ND	ND	--	ND	ND	--
SA-1484	---	ND	ND	--	ND	ND	--
CE Weld	---	ND	ND	--	ND	ND	--

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-2140
(2) Statement not required.
(3) BAW-1803, Revision 1

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1	
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements	
Plant: Surry Unit 1	
Question I: Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II: Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.	
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.	
Question III: If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response: Not applicable (see Question I and II above)	
Question IV: If plan is not to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response: Not applicable (see Question I and II above)	

NOTES: WCAP-7723: Surveillance Program Description
(ASTM E 185-66)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Surry Unit 1					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD					Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
SA-1585	70 (6)	5, approx.	44(2) 54(3)	39(2) 51(3)	
LIMITING BELTLINE PLATE OR FORGING					
C4326-1	91 (7)	>32	NA	NA	NA
C4326-2	91 (7)	>32	NA	NA	NA
C4415-1	91 (7)	>32	NA	NA	NA
C4415-2	91 (7)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.2.2. (See also letter to U. S. Nuclear Regulatory Commission from W. L. Stewart dated December 1, 1989. Title: "Virginia Electric and Power Company, Surry Unit 1 and 2: Response to Request for Additional Information, Upper-Shelf Energy of Reactor Vessel Materials." Docket Nos. 50-280 and 50-281, License Nos. DPR-32 and DPR-37)

(4) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Surry Unit 1 analysis.

(5) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(6) BAW-1803

(7) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements

Plant: Surry Unit 1

Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropwt. Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
122V109	48,55,41 42,62,47	-35	0	-15	+40	+40	NB-2331	(3,6)
PLATE								
C4326-1	28,39,47	+5	+40	+37	+10	+10	NB-2331	(1,2,7)
C4326-2	58,34,60	-5	+25	+20	0	0	NB-2331	(1,2,7)
C4415-1	40,32,34	+10	+45	+35	+20	+20	NB-2331	(1,2,7)
C4415-2	50,55,46	0	+20	+17	0	0	NB-2331	(1,2,7)
WELD								
J726	54,77,51	ND	ND	ND	ND	0	Est. (3)	(1,8)
SA-1585	31,32,31 50,54,51	ND	ND	ND	ND	-5	Est. (5)	(1,9,10)
SA-1650	48,40,40	ND	ND	ND	ND	-5	Est. (5)	(1,11)
SA-1494	54,25,44	ND	ND	ND	ND	-5	Est. (5)	(1,11)
SA-1526	33,33,33	ND	ND	ND	ND	-5	Est. (5)	(1,11)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-2150
(2) Supplier Test Report.
(3) BAW-1909, Revision 1
(4) BAW-10046P, pp 3-17, -18; mean of most conservative value for each of 24 cases.
(5) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.
(6) Values from BWNS Drawing 02-1167427E-00, Sheet 2 of 3.
(7) Values are for 60 hr stress relief.
(8) Values are for 30 hr stress relief.
(9) C_v(+10F) values; specimens from center and surface of test block.
(10) C_v(+10F) values are for 80 hr stress relief.
(11) C_v(+10F) values are for 48 hr stress relief.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>		
Plant: Surry Unit 1		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
<u>BELTLINE MATERIALS</u>		
122V109VA1	1550F-11h/WQ; 1200F-22h/AC; 1125F-40h/FC	(1)
C4326-1	1675±25F-9h/WQ; 1210F-9h/AC; 1125±25F-60h/FC	
C4326-2	1675±25F-9h/WQ; 1210F-9h/AC; 1125±25F-60h/FC	
C4415-1	1675±25F-9h/WQ; 1200-1225F-9h/AC; 1125±25F-60h/FC	
C4415-2	1675±25F-9h/WQ; 1200-1225F-9h/AC; 1125±25F-60h/FC	
J726	1130F-30h/FC	
SA-1585	1125±25F-80h/FC	
SA-1650	1125±25F-48h/FC	
SA-1494	1125±25F-48h/FC	
SA-1526	1125±25F-48h/FC	
<u>SURVEILLANCE MATERIALS</u>		
C4326-1	1650-1700F-9h/WQ; 1210F-9h/AC; 1125F-15½h/FC	(1)
C4415-1	1650-1700F-9h/WQ; 1200F-9h/AC; 1125F-15½h/FC	
SA-1526	1125F-15½h/FC	

NOTES:

(1) BAW-1909, Revision 1

(2) WQ - water quench
AC - air cool
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Surry Unit 1

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
NB Forging	122V109	NB to IS Circ.: J726	25017	1197	(1,2)
IS Plate	C4326-1	IS to LS Circ.(ID 40%): SA-1585	72445	8597	
IS Plate	C4326-2	IS to LS Circ.(OD 40%): SA-1650	72445	8632	
LS Plate	C4415-1	IS Longit.: SA-1494	8T1554	8579	
LS Plate	C4415-2	LS Longit.: SA-1494	8T1554	8579	
		LS Longit.: SA-1526	299L44	8596	

NOTES: (1) BAW-2150

(2) BAW-1909, Revision 1

(3) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Surry Unit 1				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
C4326-1 C4415-1	SA-1526	299L44	8596	(1,2)

NOTES: (1) BAW-2150
(2) BAW-1909, Revision 1

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Surry Unit 1

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
122V109	0.22	0.70	0.010	0.011	0.24	0.36	0.74	0.60	0.09	(2)
C4326-1	0.23	1.35	0.008	0.015	0.23	0.07	0.55	0.55	0.11	(2)
C4326-2	0.23	1.35	0.008	0.015	0.23	0.07	0.55	0.55	0.11	(2)
C4415-1	0.22	1.33	0.014	0.014	0.23	0.08	0.50	0.55	0.11	(2)
C4415-2	0.22	1.33	0.014	0.014	0.23	0.08	0.50	0.55	0.11	(2)
J726	0.09	1.67	ND	ND	0.27	ND	0.10	0.44	0.33	(2)
SA-1494	0.09	1.52	0.015	0.012	0.44	0.08	0.63	0.37	0.18	(3)
SA-1585	0.08	1.45	0.016	0.016	0.51	0.09	0.59	0.38	0.21	(3)
SA-1650	0.08	1.43	0.018	0.014	0.40	0.09	0.59	0.38	0.21	(3)
SA-1526	0.09	1.53	0.013	0.017	0.53	0.09	0.68	0.42	0.35	(3)
LS TO Dutchman	NA	NA	NA	NA	NA	NA	NA	NA	NA	(1)
SURVEILLANCE MATERIALS										
C4326-1	0.23	1.35	0.008	0.015	0.23	0.07	0.55	0.55	0.11	(4)
C4415-1	0.22	1.33	0.014	0.014	0.23	0.08	0.50	0.55	0.11	(4)
SA-1526	0.09	1.53	0.013	0.017	0.53	0.09	0.68	0.42	0.35	(4)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES FOR TABLE 7 ARE ON THE FOLLOWING PAGE.

TABLE 7. (CONTINUED)

NOTES:

(1) Lower shell-to-dutchman weld was fabricated by Rotterdam; information is not available.
(2) BAW-1909, Revision 1
(3) BAW-2121P
(4) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature

Plant: Surry Unit 1

Cold Leg Temperature (T_{cold}): 543 F (See Figure 4-5)

If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not applicable

References:

None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b	
Subject:	Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant:	Surry Unit 1
Were surveillance results used in determining C_vUSE? Yes ✓ No ☐	
Were surveillance results used in determining RT_{NDT}? Yes ☐ No ✓	
If any "yes" boxes were checked above, state how the surveillance results were used: Determination of Upper-Shelf Energy per Regulatory Guide 1.99, Revision 2, Position 2, for SA-1585 weld materials only.	
References:	Letter to U. S. Nuclear Regulatory Commission from W. L. Stewart dated December 1, 1989. Title: "Virginia Electric and Power Company, Surry Unit 1 and 2: Response to Request for Additional Information, Upper-Shelf Energy of Reactor Vessel Materials." Docket Nos. 50-280 and 50-281, License Nos. DPR-32 and DPR-37.

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Surry Unit 1

Question I. Does measured ΔRT_{NDT} exceed $\Delta RT_{NDT} + 2\sigma$ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I yes/no	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II yes/no
122V109	NA	ND	ND	NA	ND	ND	NA
C4326-1	NA	ND	ND	NA	ND	ND	NA
C4326-2	NA	ND	ND	NA	ND	ND	NA
C4415-1	2.86E+18(2)	50(1)	82	No	5(1)	19	No
	1.94E+19(1)	110(1)	120	No	9(1)	29	No
C4415-2	NA	ND	ND	NA	ND	ND	NA
J726	NA	ND	ND	NA	ND	ND	NA
SA-1585	5.10E+18(2)	148(2)	188	No	22(2)	24(3)	No
SA-1650	NA	ND	ND	NA	ND	ND	NA
SA-1494	NA	ND	ND	NA	ND	ND	NA
SA-1526	2.86E+18(2)	167(2)	203	No	17(2)	25	No
	1.94E+19(1)	240(1)	320	No	20(1)	33	No

NOTES:
(1) WCAP-11415
(2) BAW-1803, Revision 1
(3) BAW-1910P

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1	
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements	
Plant: Surry Unit 2	
Question I: Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II: Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.	
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.	
Question III: If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response: Not applicable (see Question I and II above)	
Question IV: If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response: Not applicable (see Question I and II above)	

NOTES: HCAP-8085: Surveillance Program Description
(ASTM E 185-70)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Surry Unit 2					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD SA-1585	70 (5)	32, approx.	NA	NA	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
LIMITING BELTLINE PLATE OR FORGING C4208-2	91 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Surry Unit 2 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)									
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements									
Plant: Surry Unit 2									
Column 1	Column 2				Column 3	Column 4	Column 5	C.6	
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropwt. Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes	
	Col. 2a	Col. 2b	Col. 2c	Col. 2d					
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F					
FORGING									
123V303	142,83,122 110,90,168 105	-20	0	+5	+30	+30	NB-2331	(3,6)	
PLATE									
C4208-2	64,68,75	-45	-20	-20	-30	-30	NB-2331	(1,2,7)	
C4339-1	ND	+25	+50	+45	-10	-10	NB-2331	(1,2,7)	
C4331-2	46,60,25	+5	+35	+32	-10	-10	NB-2331	(1,2,7)	
C4339-2	48,45,25	-5	+25	+10	-20	-20	NB-2331	(1,2,7)	
WELD									
L737	75,62,66	ND	ND	ND	ND	0	Est. (3)	(1,8)	
SA-1585	31,32,31 50,54,51	ND	ND	ND	ND	-5	Est. (5)	(1,9,10)	
R3008	66,51,46	ND	ND	ND	ND	0	Est. (3)	(1,11)	
WF-4	40,31,34	ND	ND	ND	ND	-5	Est. (5)	(1,10)	
WF-8	45,38,30	ND	ND	ND	ND	-5	Est. (5)	(1,12)	

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-2150
(2) Supplier Test Report.
(3) BAW-1909, Revision 1
(4) BAW-10046P, pp 3-17, -18; mean of most conservative value for each of 24 cases.
(5) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.
(6) Values from BWNS Drawing 02-1167428E-00, Sheet 2 of 3.
(7) Values are for 60 hr stress relief.
(8) Values are for 24 hr stress relief.
(9) C_v(+10F) values; specimens from center and surface of test block.
(10) C_v(+10F) values are for 80 hr stress relief.
(11) C_v(+10F) values are for 25 hr stress relief.
(12) C_v(+10F) values are for 48 hr stress relief.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>		
Plant: Surry Unit 2		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
<u>BELTLINE MATERIALS</u>		
123V303VA1	1550F-12h/WQ; 1200F-22h/AC; 1125F-40h/FC	(1)
C4331-2	1600-1650F-9h/BQ; 1200-1225F-9h/BQ; 1125F-60h/FC	
C4339-2	1600-1650F-9h/BQ; 1200-1225F-9h/BQ; 1125F-60h/FC	
C4339-1	1600-1650F-9h/BQ; 1200-1225F-9h/BQ; 1125F-60h/FC	
C4208-2	1600-1650F-9h/BQ; 1200-1225F-9h/BQ; 1125F-60h/FC	
L737	1130F-24h/FC	
R3008	1130F-25h/FC	
SA-1585	1125±25F-80h/FC	
WF-4	1125±25F-80h/FC	
WF-8	1125±25F-48h/FC	
<u>SURVEILLANCE MATERIALS</u>		
C4339-1	1625F-9h/BQ; 1212F-9h/BQ; 1140F-15½h/FC	(1)
R3008	1140F-15½h/FC	

NOTES:

(1) BAW-1909, Revision 1
(2) BQ - brine quench
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Surry Unit 2

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
NB Forging	123V303	NB to IS Circ.: L737	4275	02275	(1,2, 3)
IS Plate	C4331-2	IS to LS Circ.: R3008	0227	LW320	
IS Plate	C4339-2	IS Longit.(ID 50%): WF-4	8T1762	8597	
LS Plate	C4208-2	IS Longit.(OD 50%): SA-1585	72445	8579	
LS Plate	C4339-1	LS Longit.: WF-4	8T1762	8597	
		LS Longit.(ID 63%): WF-4	8T1762	8597	
		LS Longit.(OD 37%): WF-8	8T1762	8632	

NOTES: (1) BAW-2150
(2) BAW-1909, Revision 1
(3) Mt. Vernon fabrication process sheets
(4) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Surry Unit 2				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
C4339-1	R3008	0227	LW320	(1,2)

NOTES: (1) BAW-2150
(2) BAW-1909, Revision 1

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>

Plant: Surry Unit 2

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
<u>BELTLINE MATERIALS</u>										
123V303	0.20	0.63	0.010	0.010	0.24	0.36	0.73	0.58	0.09	(2)
C4331-2	0.23	1.42	0.009	0.015	0.22	ND	0.60	0.55	0.12	(2)
C4339-2	0.23	1.30	0.012	0.014	0.25	ND	0.54	0.54	0.11	(2)
C4208-2	0.21	1.28	0.008	0.013	0.24	ND	0.55	0.55	0.15	(2)
C4339-1	0.23	1.30	0.012	0.014	0.25	ND	0.54	0.54	0.11	(2)
L737	0.08	1.74	ND	ND	0.35	ND	0.10	0.38	0.35	(2)
SA-1585	0.08	1.45	0.016	0.016	0.51	0.09	0.59	0.38	0.21	(3)
R3008	0.09	1.51	0.017	0.016	0.46	0.10	0.56	0.41	0.19	(2)
WF-4	0.07	1.48	0.017	0.011	0.51	0.12	0.55	0.41	0.20	(3)
WF-8	0.06	1.45	0.009	0.009	0.53	0.12	0.55	0.41	0.20	(3)
LS-to-Dutchman	NA	NA	NA	NA	NA	NA	NA	NA	NA	(1)
<u>SURVEILLANCE MATERIALS</u>										
C4339-1	0.23	1.30	0.012	0.014	0.25	0.08	0.54	0.54	0.11	(4)
R3008	0.09	1.51	0.017	0.016	0.46	0.10	0.56	0.41	0.19	(4)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES FOR TABLE 7 ARE ON THE FOLLOWING PAGE.

TABLE 7. (CONTINUED)

NOTES:

(1) Lower shell-to-dutchman weld was fabricated by Rotterdam; information is not available (NA).
(2) BAW-1909, Revision 1
(3) BAW-2121P
(4) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Surry Unit 2
Cold Leg Temperature (T_{cold}): 543 F (See Figure 4-5)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2: Not applicable
References: None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b		
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results		
Plant: Surry Unit 2		
Were surveillance results used in determining C_vUSE?	Yes ☐	No ✓
Were surveillance results used in determining RT_{NDT}?	Yes ☐	No ✓
If any "yes" boxes were checked above, state how the surveillance results were used:		
References: None		

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Surry Unit 2

Question I. Does measured ΔRT_{NDT} exceed $\Delta RT_{NDT} + 2\sigma$ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I Yes/No	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II yes/no
123V303	NA	ND	ND	NA	ND	ND	NA
C4208-2	NA	ND	ND	NA	ND	ND	NA
C4339-1	3.02E+18(1)	45(1)	83	No	10(1)	16	No
	1.88E+19(1)	75(1)	120	No	11(1)	24	No
C4331-2	NA	ND	ND	NA	ND	ND	NA
C4339-2	NA	ND	ND	NA	ND	ND	NA
L737	NA	ND	ND	NA	ND	ND	NA
R3008	3.02E+18(1)	95(1)	157	No	20(1)	22	No
	1.88E+19(1)	145(1)	233	No	30(1)	34	No
SA-1585	5.10E+18(2)	148(2)	188	No	22(2)	24(3)	No
WF-4	NA	ND	ND	NA	ND	ND	NA
WF-8	NA	ND	ND	NA	ND	ND	NA

NOTES:
(1) WCAP-11499
(2) BAW-180[illegible] Revision 1
(3) BAW-19

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Three Mile Island Unit 1		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: BAW-10006A, Revision 3: Surveillance Program Description (ASTM E 185-70)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Three Mile Island Unit 1					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	End of License (26.17)	
LIMITING BELTLINE WELD WF-25	70 (5)	4, approx.	47	44	Analysis per 10CFR50, Appendix G, Section V.C.3. is scheduled for 1993 under the sponsorship of B&WOG Reactor Vessel Working Group.
LIMITING BELTLINE PLATE OR FORGING C3307-1	91 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES:
(1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and End of License (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) Analyses that have demonstrated the required margin of safety for the Zion and Turkey Point plants at load level A and B conditions have been submitted to the NRC (Reports BAW-2118P and BAW-2148P). The results of these two analyses are anticipated to bound the outcome of the Three Mile Island Unit 1 analysis.

(4) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(5) BAW-1803

(6) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Three Mile Island Unit 1								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
ARY 59	117,110,101 120,122,123	ND	ND	ND	ND	+3	Est. (2)	(1,6)
PLATE								
C2789-1	50,36,33	ND	ND	ND	0	+1	Est. (3)	(1,7)
C2789-2	42,37,35	ND	ND	ND	+10	+1	Est. (3)	(1,7)
C3307-1	ND	ND	ND	ND	+10	+1	Est. (3)	(1,7)
C3251-1	43,40,29 71,59,26	ND	ND	ND	-10	+1	Est. (3)	(1,7)
WELD								
WF-70	39,35,44	ND	ND	ND	ND	+18	Eval.(4)	(1,7,9)
WF-25	38,28,49	ND	ND	ND	ND	-5	Est. (5)	(1,8)
WF-67	29,35,30	ND	ND	ND	ND	-5	Est. (5)	(1,8)
WF-8	45,38,30	ND	ND	ND	ND	-5	Est. (5)	(1,8)
SA-1526	33,33,33	ND	ND	ND	ND	-5	Est. (5)	(1,8)
SA-1494	54,25,44	ND	ND	ND	ND	-5	Est. (5)	(1,8)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-1820

(2) BAW-10046P, pp 3-17, -18; mean of most conservative value for each of 24 cases.

(3) BAW-10046P, pp 3-18; mean of most conservative value for each of 13 cases.

(4) BAW-2100

(5) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(6) C_v(+10F) values are for 60 hr stress-relief.

(7) C_v(+10F) values are for 40 hr stress-relief.

(8) C_v(+10F) values are for 48 hr stress-relief.

(9) RT_{NDT} value for 40 hr stress-relief maximum.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: Three Mile Island Unit 1		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
ARY 59	1600±20F-7h/WQ; 1230±20F-14h/WQ; 1100-1150F-45¼h/FC (cumul.)	(1,2)
C2789-1	1510-1535F-5h/BQ; 1200-1225F-5h/BQ; 1100-1150F-36h/FC (cumul.)	
C2789-2	1510-1535F-5h/BQ; 1200-1225F-5h/BQ; 1100-1150F-36h/FC (cumul.)	
C3307-1	1600-1650F-9½h/BQ; 1200-1225F-9½h/BQ; 1225-1250F-9½h/BQ; 1100-1150F-37¼h/FC (cumul.)	
C3251-1	1600-1650F-9½h/BQ; 1200-1225F-9½h/BQ; 1100-1150F-37¼h/FC (cumul.)	
WF-70	1100-1150F-27½h/FC	
WF-25	1100-1150F-35h/FC (cumul.)	
WF-67/WF-70	1100-1150F-35h/FC (cumul.)	
WF-8	1100-1150F-36h/FC (cumul.)	
SA-1526/SA-1494	1100-1150F-37¼h/FC (cumul.)	
SURVEILLANCE MATERIALS		
C2789-2	1510-1535F-5h/BQ; 1200-1225F-5h/BQ; 1100-1150F-27½h/FC	(1)
C3307-1	1600-1650F-9½h/BQ; 1200-1225F-9½h/BQ; 1225-1250F-9½h/BQ; 1100-1150F-27½h/FC	
WF-25	1100-1150F-27½h/FC	

NOTES FOR TABLE 4 ARE ON FOLLOWING PAGE.

TABLE 4. (CONTINUED)

NOTES:

(1) BAW-1820
(2) Additional stress relief information per Mt. Vernon process drawing.
(3) WQ - water quench
BQ - brine quench
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Three Mile Island Unit 1

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Lower NB Forging	ARY 59, 123S454	NB to US Circ.: WF-70	72105	8669	(1)
US Plate	C2789-1	US to LS Circ.: WF-25	299L44	8650	
US Plate	C2789-2	LS to Dutch Circ.(ID 50%): WF-67	72442	8669	
LS Plate	C3307-1	LS to Dutch Circ.(OD 50%): WF-70	72105	8669	
LS Plate	C3251-1	US Longit.: WF-8	8T1762	8632	
		LS Longit.: SA-1526	299L44	8596	
		LS Longit.: SA-1494	8T1554	8579	

NOTES: (1) BAW-1820
(2) NB - Nozzle Belt
US - Upper Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Three Mile Island Unit 1				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
C2789-2 C3307-1	WF-25	299L44	8650	(1)

NOTES: (1) BAW-1820

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Three Mile Island Unit 1

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
ARY 59	0.26	0.63	0.006	0.008	0.28	0.34	0.72	0.64	0.08	(1)
C2789-1	0.24	1.36	0.010	0.017	0.23	0.19	0.57	0.51	0.09	(1)
C2789-2	0.24	1.36	0.010	0.017	0.23	0.19	0.57	0.51	0.09	(1)
C3307-1	0.21	1.24	0.010	0.016	0.27	0.12	0.55	0.47	0.12	(1)
C3251-1	0.23	1.41	0.012	0.013	0.21	0.14	0.50	0.47	0.11	(1)
SA-1494	0.09	1.52	0.015	0.012	0.44	0.08	0.63	0.37	0.18	(2)
SA-1526	0.09	1.53	0.013	0.017	0.53	0.09	0.68	0.42	0.35	(2)
WF-8	0.06	1.45	0.009	0.009	0.53	0.12	0.55	0.41	0.20	(2)
WF-25	0.09	1.60	0.015	0.016	0.50	0.09	0.68	0.42	0.35	(2)
WF-67	0.08	1.55	0.021	0.016	0.58	0.09	0.60	0.39	0.24	(2)
WF-70	0.09	1.63	0.018	0.009	0.54	0.10	0.59	0.40	0.35	(2)
SURVEILLANCE MATERIALS										
C2789-2	0.24	1.36	0.010	0.017	0.23	0.19	0.57	0.51	0.09	(3)
C3307-1	0.21	1.24	0.010	0.016	0.27	0.12	0.55	0.47	0.12	(3)
WF-25	0.09	1.62	0.014	0.015	0.46	0.10	0.66	0.40	0.33	(3)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES FOR TABLE 7 ARE ON THE FOLLOWING PAGE.

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Three Mile Island Unit 1
Were surveillance results used in determining C_vUSE? Yes ☐ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ☐ No ✓
If any "yes" boxes were checked above, state how the surveillance results were used:
References: None

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Three Mile Island Unit 1

Question I. Does measured ΔRT_{NDT} exceed $\Delta RT_{NDT} + 2\sigma$ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (2,4)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (5)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (5)
ARY 59	---	ND	ND	--	ND	ND	--
C2789-1	---	ND	ND	--	ND	ND	--
C2789-2	1.07E+18	5(1)	50	No	10(1)	10(1)	No
	8.66E+18	13(1)	90	No	19(1)	17(1)	Yes
C3307-1	---	ND	ND	--	ND	ND	--
C3251-1	---	ND	ND	--	ND	ND	--
WF-70	6.63E+18	135(2)	259	No	13(2)	22(6)	No
WF-25	1.07E+18	124(2)	148	No	17(2)	24(1)	No
	8.66E+18	203(2)	261	No	31(2)	34(1)	No
	7.79E+18	214(2)	263	No	25(2)	30(7)	No
WF-67	6.09E+18	160(2)	200	No	15(2)	23(7)	No
WF-8	---	ND	ND	--	ND	ND	--
SA-1526	2.86E+18	167(2)	203	No	17(2)	25	No
	1.94E+19	240(3)	320	No	20(3)	33	No
SA-1494	---	ND	ND	--	ND	ND	--
Atypical	1.17E+18	28(4)	138	No	9(4)	25(4)	No
	6.56E+18	122(4)	216	No	16(4)	32(4)	No
	7.50E+18	119(4)	223	No	11(4)	32(4)	No
	1.08E+19	120(4)	242	No	15(4)	34(4)	No

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 7. (CONTINUED)

NOTES:

(1) BAW-1820

(2) BAW-2:21P

(3) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Three Mile Island Unit 1
Cold Leg Temperature (T_{cold}): 556 F (See Figure 4-1)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2: Not applicable
References: None

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-1901
(2) BAW-1803, Revision 1
(3) WCAP-11415
(4) BAW-2049
(5) The only instance where a measured "shift" or "drop" exceeds that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2, is for a "drop" for base metal and the predicted "drop" exceeds the measured "drop" by 2 ft-lbs which is not considered significant. The requirements of 10CFR50, Appendix G, were not violated, and there being no further application of the "drop" data, the effect of these surveillance results are therefore not significant.
(6) BAW-1920P
(7) BAW-1910P

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Turkey Point Unit 3		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: WCAP-7656: Surveillance Program Description
(ASTM E 185-66)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Turkey Point Unit 3					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD SA-1101	65 (4)	2, approx.	39	36	An analysis which demonstrates that this material provides margin of safety against fracture equivalent to that required by ASME Section III, Appendix G, has been performed under the sponsorship of the B&W Owners Group's Reactor Vessel Working Group and submitted to the NRC as report BAW-2118P.
LIMITING BELTLINE PLATE OR FORGING 123S266	154 (4)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(4) BAW-2150; Based on TP-3 surveillance material test data.

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, IT"M b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Turkey Point Unit 3								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
<u>FORGING</u>								
122S146	73,52,62 97,65,88	0	+15	+15	+50	+50	NB-2331	(2,4)
123P461	99,63,86 78,83,84	-28	-10	-16	+40	+40	NB-2331	(1,2,4)
123S266	88,38,87 93,130,89	-48	-27	-31	+30	+30	NB-2331	(1,2,4)
<u>WELD</u>								
SA-1484	40,52,41	ND	ND	ND	ND	-5	Est. (3)	(1)
SA-1101	45,45,46	ND	+70	ND	-70	+10	NB-2331	(1,5,6)
SA-1135	56,44,55	ND	ND	ND	ND	-5	Est. (3)	(1)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-2150

(2) Supplier Test Report

(3) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(4) Values are for 40 hr stress-relief.

(5) C_v(+10F) values are for 8 - 6 hr cycles stress relief.

(6) EPRI NP-373; C_v 50 ft-lb, Drop Weight, and RT_{NDT} values.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Turkey Point Unit 3

Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		(1,2)
122S146VA1	1550F-11h/WQ; 1220F-22h/AC; 1125F-11h (min)/FC	
123P461VA1	1550F-13h/WQ; 1210F-18h/AC; 1125F-9¾h (min)/FC	
123S266VA1	1550F-13h/WQ; 1210F-18h/AC; 1125F-9¾h (min)/FC	
SA-1484	1125F-9¼h (min)/FC	
SA-1101	1125F-9½h (min)/FC	
SA-1135	1125F-9½h (min)/FC	
SURVEILLANCE MATERIALS		(1)
123P461VA1	1550F-13h/WQ; 1210F-8h/AC; 1125F-10½h/FC	
123S266VA1	1550F-13h/WQ; 1210F-8h/AC; 1125F-10½h/FC	
SA-1101	1125F-10¼h/FC	

NOTES:

(1) BAW-2150
(2) Additional stress relief information per Mt. Vernon fabrication process sheets.
(3) WQ - water quench
AC - air cool
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA					
Plant: Turkey Point Unit 3					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
NB Forging	122S146	NB to IS Circ.: SA-1484	72442	8579	(1,2)
IS Forging	123P461	IS to LS Circ.: SA-1101	71249	8445	
LS Forging	123S266	LS to Dutch Circ.: SA-1135	61782	8457	

NOTES: (1) BAW-2150
(2) Mt. Vernon fabrication process sheets
(3) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Turkey Point Unit 3				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
123P461 123S266	SA-1101	71249	8445	(1)

NOTES: (1) BAW-2150

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>

Plant: Turkey Point Unit 3

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
<u>BELTLINE MATERIALS</u>										
122S146VA1	0.22	0.64	0.010	0.013	0.25	0.34	0.68	0.58	ND	(1)
123P461VA1	0.20	0.64	0.010	0.010	0.26	0.40	0.70	0.62	0.06	(2,3)
123S266VA1	0.20	0.62	0.010	0.008	0.20	0.38	0.67	0.58	0.08	(2,3)
SA-1484	0.08	1.52	0.018	0.015	0.42	0.09	0.60	0.39	0.24	(4)
SA-1101	0.07	1.28	0.021	0.014	0.52	0.16	0.60	0.37	0.26	(4)
SA-1135	0.08	1.45	0.011	0.013	0.49	0.08	0.54	0.38	0.25	(4)
<u>SURVEILLANCE MATERIALS</u>										
123P461VA1	0.20	0.64	0.010	0.010	0.26	0.40	0.70	0.62	0.06	(5)
123S266VA1	0.20	0.62	0.010	0.008	0.20	0.38	0.67	0.58	0.08	(5)
SA-1101	0.08	1.51	0.020	0.013	0.57	0.16	0.60	0.37	0.26	(5)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) Supplier Material Test Report
(2) BAW-2150
(3) Copper content based on surveillance material data.
(4) BAW-2121P
(5) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature

Plant: Turkey Point Unit 3

Cold Leg Temperature (T_{cold}): 546 F (See Figure 4-6)

If T_{cold} is <525 F, state how this is considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not Applicable

References:

None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b

Subject:	Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant:	Turkey Point Unit 3
Were surveillance results used in determining C_vUSE?	Yes ☐ No ✓
Were surveillance results used in determining RT_{NDT}?	Yes ✓ No ☐

If any "yes" boxes were checked above, state how the surveillance results were used:

Turkey Point Units 3 and 4 - Issuance of Amendments RE: Pressure and Temperature (P/T) Limits (TAC Nos. 69390 and 69391).

References: Letter to W. F. Conway from G. E. Edison dated January 10, 1989.

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c							
Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects							
Plant: Turkey Point Unit 3							
Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?							
Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?							
Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (6)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (6)
122S146	---	ND	ND	--	ND	ND	--
123P461	7.01E+18(4)	0(1,5)	67	No	0(1)	20	No
	1.41E+19(2)	23(2,5)	75	No	17(2)	24	No
123S266	1.41E+19(2)	45(2,5)	90	No	32(2)	29	Yes
	1.23E+19(4)	45(3)	88	No	0(3)	28	No
SA-1484	---	ND	ND	--	ND	ND	--
SA-1101	7.01E+18(4)	164(4)	195	No	4(4)	21	No
	1.23E+19(4)	178(4)	220	No	18(4)	24	No
SA-1135	1.03E+19(4)	142(4)	240	No	21(4)	31(7)	No

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) WCAP-8631
(2) SWRI 02-5131 and SWRI 02-5380
(3) SWRI 06-8575
(4) BAW-1803, Revision 1
(5) 50 ft-lb transition temperature
(6) The only instance where a measured "shift" or "drop" exceeds that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2, is for a "drop" for base metal. The requirements of 10CFR50, Appendix G, were not violated, and there being no further application of the "drop" data, the effect of these surveillance results are therefore not significant.
(7) BAW-1920P

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Turkey Point Unit 4		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: WCAP-7660: Surveillance Program Description
(ASTM E 185-66)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Turkey Point Unit 4					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
<u>LIMITING BELTLINE WELD</u> SA-1101	65 (4)	2, approx.	40	36	An analysis which demonstrates that this material provides margin of safety against fracture equivalent to that required by ASME Section III, Appendix G, has been performed under the sponsorship of the B&W Owners Group's Reactor Vessel Working Group and submitted to the NRC as report BAW-2118P.
<u>LIMITING BELTLINE PLATE OR FORGING</u> 122S180	132 (4)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(4) BAW-2150; Based on TP-3 surveillance test data.

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Turkey Point Unit 4								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
124S309	80,101,89 49,74,88,64	0	+20	+18	+40	+40	NB-2331	(2,5)
123P481	44,62,28 30,58,46	+15	+45	+40	+50	+50	NB-2331	(1,2,5)
123S180	91,59,64 62,56,53	-37	-15	-25	+40	+40	NB-2331	(1,2,5)
WELD								
WF-67	29,35,30	ND	ND	ND	ND	-5	Est. (3)	(1,6)
WF-70	39,35,44	ND	ND	ND	ND	+18	Eval.(4)	(1,6,9)
SA-1101	45,45,46	ND	+70	ND	-70	+10	NB-2331	(1,7,8)
SA-1135	56,44,55	ND	ND	ND	ND	-5	Est. (3)	(1)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-2150

(2) Supplier Test Report

(3) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(4) BAW-2100

(5) Values are for 40 hr stress-relief.

(6) C_v(+10F) values are for 48 hr stress-relief.

(7) C_v(+10F) values are for 8 - 6 hr cycles stress-relief.

(8) EPRI NP-373; C_v 50 ft-lb, Drop Weight, and RT_{NDT} values.

(9) RT_{NDT} value for 40 hr stress-relief maximum.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>		
Plant: Turkey Point Unit 4		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
<u>BELTLINE MATERIALS</u>		
124S309VA1	1550F-15h/WQ; 1220F-22h/FC; 1125F-10½h/FC	(1,2)
123P481VA1	1550F-10½h/WQ; 1210F-18h/AC; 1125F-10¾h (min)/FC	
122S180VA1	1550F-10½h/WQ; 1200F-18h/FC; 1125F-10¾h (min)/FC	
WF-67/WF-70	1125F-9½h (min)/FC	
SA-1101	1125F-10½h (min)/FC	
SA-1135	1125F-10½h (min)/FC	
<u>SURVEILLANCE MATERIALS</u>		
123P481VA1	1550F-10½h/WQ; 1200F-18h/AC; 1125F-10½h/FC	(1)
122S180VA1	1550F-10½h/WQ; 1210F-18h/AC; 1125F-10½h/FC	
SA-1094	1125F-10½h/FC	

NOTES:

(1) BAW-2150
(2) Additional stress relief information per Mt. Vernon fabrication process sheets.
(3) WQ - water quench
AC - air cool
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA					
Plant: Turkey Point Unit 4					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
NB Forging	124S309	NB to IS Circ.(ID 67%): WF-67	72442	8669	(1,2)
IS Forging	123P481	NB to IS Circ.(OD 33%): WF-70	72105	8669	
LS Forging	123S180	IS to LS Circ.: SA-1101	71249	8445	
		LS to Dutch Circ.: SA-1135	61782	8457	

NOTES: (1) BAW-2150
(2) Mt. Vernon fabrication process sheets
(3) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>				
Plant: Turkey Point Unit 4				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
123P481 122S180	SA-1094	71249	8457	(1)

NOTES: (1) BAW-2150

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)										
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA										
Plant: Turkey Point Unit 4										
Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
124S309VA1	0.20	0.60	0.010	0.012	0.26	0.33	0.70	0.56	ND	(1)
123P481VA1	0.20	0.65	0.010	0.010	0.24	0.32	0.68	0.59	0.05	(2)
122S180VA1	0.22	0.60	0.010	0.009	0.22	0.34	0.74	0.60	0.06	(2)
WF-70	0.09	1.63	0.018	0.009	0.54	0.10	0.59	0.40	0.35	(3)
WF-67	0.08	1.55	0.021	0.016	0.58	0.09	0.60	0.39	0.24	(3)
SA-1101	0.07	1.28	0.021	0.014	0.52	0.16	0.60	0.37	0.26	(3)
SA-1135	0.08	1.45	0.011	0.013	0.49	0.08	0.54	0.38	0.25	(3)
SURVEILLANCE MATERIALS										
123P481VA1	0.22	0.67	0.010	0.009	0.20	0.33	0.71	0.56	0.05	(4)
122S180VA1	0.21	0.67	0.011	0.009	0.23	0.31	0.79	0.56	0.06	(4)
SA-1094	0.10	1.44	0.014	0.011	0.50	0.14	0.60	0.36	0.26	(4)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) Supplier Material Test Report.
(2) BAW-2150
(3) BAW-2121P
(4) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature

Plant: Turkey Point Unit 4

Cold Leg Temperature (T_{cold}): 546 F (See Figure 4-6)

If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not applicable

References:

None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Turkey Point Unit 4
Were surveillance results used in determining C_vUSE? Yes □ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No □
If any "yes" boxes were checked above, state how the surveillance results were used: Turkey Point Units 3 and 4 - Issuance of Amedendments RE: Pressure and Temperature (P/T) Limits (TAC Nos. 69390 and 69391)
References: Letter to W. F. Conway from G. E. Edison dated January 10, 1989.

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Turkey Point Unit 4

Question I. Does measured ΔRT_{NDT} exceed $\Delta RT_{NDT} + 2\sigma$ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (5)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (5)
124S309	---	ND	ND	--	ND	ND	--
123P481	1.25E+19(1)	35(1,4)	66	No	12(1)	20	No
123S180	7.54E+18(3)	10(2,4)	68	No	0(2)	18	No
	1.25E+19(1)	11(1,4)	73	No	10(1)	21	No
WF-67	6.09E+18(3)	160(3)	200	No	15(3)	23(6)	No
WF-70	6.63E+18(3)	135(3)	259	No	13(3)	22(7)	No
SA-1101	7.01E+18(3)	164(3)	195	No	4(3)	21	No
	1.23E+19(3)	178(3)	220	No	18(3)	24	No
SA-1135	1.03E+19(3)	142(3)	240	No	21(3)	31(7)	No

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) SWRI 02-5131 and SWRI 02-5380
(2) SWRI 02-4221
(3) BAW-1803, Revision 1
(4) 50 ft-lb transition temperature
(5) Statement not required.
(6) BAW-1910P
(7) BAW-1920P

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1	
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements	
Plant: Zion Unit 1	
Question I: Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ☐ No ✓
Question II: Is plant one of the following? ANO ., Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No ☐
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.	
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.	
Question III: If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response: Not applicable (see Question I and II above)	
Question IV: If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response: Not applicable (see Question I and II above)	

NOTES: WCAP-8064: Surveillance Program Description
(ASTM E 185-70)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Zion Unit 1					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD WF-70	70 (4)	2, approx.	44	40	An analysis which demonstrates that this material provides margin of safety against fracture equivalent to that required by ASME Section III, Appendix G, has been performed under the sponsorship of the B&W Owners Group's Reactor Vessel Working Group and submitted to the NRC as report BAW-2148P.
LIMITING BELTLINE PLATE OR FORGING B7823-1	91 (5)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(4) BAW-1803

(5) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Zion Unit 1								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
ANA 102	ND	+15	+45	+35	+20	+20	NB-2331	(2)
PLATE								
C3795-2	42,44,39	-10	+25	+15	-10	-10	NB-2331	(1,2,5)
B7835-1	ND	+15	+32	+25	-20	-20	NB-2331	(1,2,5)
C3799-2	40,35,43	-2	+33	+25	-20	-20	NB-2331	(1,2,5)
B7823-1	27,40,33	+5	+27	+20	-20	-20	NB-2331	(1,2,5)
WELD								
WF-154	41,37,43	ND	ND	ND	ND	-5	Est. (3)	(1,6)
SA-1769	36,35,38	ND	ND	ND	ND	-5	Est. (3)	(1,7)
WF-70	39,35,44	ND	ND	ND	ND	+18	Eval.(4)	(1,6,9)
WF-4	40,31,34	ND	ND	ND	ND	-5	Est. (3)	(1,8)
WF-8	45,38,30	ND	ND	ND	ND	-5	Est. (3)	(1,6)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-2150

(2) Mt. Vernon Qualification Test Report

(3) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(4) BAW-2100

(5) Values are for 60 hr stress-relief.

(6) C_v(+10F) values are for 48 hr stress-relief.

(7) C_v(+10F) values are for 8 - 6 hr cycles stress-relief.

(8) C_v(+10F) values are for 80 hr stress-relief.

(9) RT_{NDT} value for 40 hr stress-relief maximum.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: Zion Unit 1		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
ANA 102	Not available	(1,2)
C3795-2	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1125F-26¼h/FC (cumul.)	
B7835-1	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1125F-26¼h/FC (cumul.)	
C3799-2	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1125F-23¼h/FC (cumul.)	
B7823-1	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1125F-23¼h/FC (cumul.)	
WF-154/SA-1769	1100-1150F-18¾h/FC (cumul.)	
WF-70	1100-1150F-23h/FC (cumul.)	
WF-154	1100-1150F-18¾h/FC (cumul.)	
WF-4/WF-8	1100-1150F-26¼h/FC (cumul.)	
WF-8	1100-1150F-23¼h/FC (cumul.)	
SURVEILLANCE MATERIALS		
B7835-1	1625F-9¾h/BQ; 1212F-9¾/h/BQ; 1125F-25h/FC	(1)
WF-209-1	1125F-23h/FC	

NOTES:

(1) BAW-2150
(2) Additional stress relief information per Mt. Vernon process drawing.
(3) BQ - brine quench
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>

Plant: Zion Unit 1

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Lower NB Forging	ANA 102	NB to IS Circ.(ID 82%): WF-154	406L44	8720	(1,2)
IS Plate	B7835-1	NB to IS Circ.(OD 18%): SA-1769	71249	8738	
IS Plate	C3795-2	IS to LS Circ.: WF-70	72105	8669	
LS Plate	B7823-1	LS to Dutch Circ.: WF-154	406L44	8720	
LS Plate	C3799-2	IS Longit.: WF-4	8T1762	8597	
		IS Longit.(ID 39%): WF-8	8T1762	8632	
		IS Longit.(OD 61%): WF-4	8T1762	8597	
		LS Longit.: WF-8	8T1762	8632	

NOTES: (1) BAW-2150
(2) Mt. Vernon process drawing
(3) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Zion Unit 1				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
B7835-1	WF-209-1	72105	8773	(1)

NOTES: (1) BAW-2150

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)										
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA										
Plant: Zion Unit 1										
Column 1	Column 2									C.3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
ANA 102	X	0.74	X	0.012	0.26	0.45	X	0.60	0.06	(1)
C3795-2	0.21	1.50	0.010	0.015	0.23	ND	0.49	0.49	0.12	(1,2)
B7835-1	[illegible]	1.30	0.010	0.011	0.20	ND	0.49	0.47	0.12	(1,2)
C3799-2	[illegible]	1.33	0.010	0.014	0.24	ND	0.50	0.46	0.15	(1,2)
B7823-1	[illegible]	1.36	0.013	0.016	0.21	ND	0.48	0.46	0.13	(1,2)
WF-154	0.07	1.54	0.013	0.016	0.42	0.07	0.59	0.40	0.31	(3)
SA-1769	0.09	1.49	0.020	0.014	0.56	0.16	0.61	0.37	0.26	(3)
WF-4	0.07	1.48	0.017	0.011	0.51	0.12	0.55	0.41	0.20	(3)
WF-8	0.06	1.45	0.009	0.009	0.53	0.12	0.55	0.41	0.20	(3)
WF-70	0.09	1.63	0.018	0.009	0.54	0.10	0.59	0.40	0.35	(3)
SURVEILLANCE MATERIALS										
B7835-1	0.20	1.30	0.010	0.011	0.20	ND	0.49	0.47	0.11	(4)
WF-209-1	0.09	1.51	0.020	0.013	0.68	0.06	0.57	0.39	0.35	(4)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) Supplier Material Test Report (X - chemical contents are not legible)
(2) BAW-2150
(3) BAW-2121P
(4) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: Zion Unit 1
Cold Leg Temperature (T_{cold}): 529.4 F (See Figure 4-7)
If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2: Not applicable
References: None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Zion Unit 1
Were surveillance results used in determining C_vUSE? Yes □ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No □
If any "yes" boxes were checked above, state how the surveillance results were used: Initial RT_{NDT} value for WF 70 weld metal.
References: BAW-2100

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: Zion Unit 1

Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured C_vUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (1,2,3)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (5)	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II If "yes" see Note (5)
ANA 102	---	ND	ND	--	ND	ND	--
C3795-2	---	ND	ND	--	ND	ND	--
B7835-1	2.53E+18	25(1)	84	No	3(1)	18(1)	No
	8.49E+18	60(1)	111	No	15(1)	24(1)	No
	1.26E+19	80(1)	112	No	24(1)	26(1)	No
	1.56E+19	94(1)	116	No	21(1)	27(1)	No
C3799-2	---	ND	ND	--	ND	ND	--
B7823-1	---	ND	ND	--	ND	ND	--
WF-154	---	ND	ND	--	ND	ND	--
SA-1769	---	ND	ND	--	ND	ND	--
WF-70	6.63E+18	135(2)	242	No	13(2)	22(4)	No
WF-4	---	ND	ND	--	ND	ND	--
WF-8	---	ND	ND	--	ND	ND	--
Atypical	1.17E+18	28(3)	138	No	9(3)	25(3)	No
	6.56E+18	122(3)	216	No	16(3)	32(3)	No
	7.50E+18	119(3)	223	No	11(3)	32(3)	No
	1.08E+19	120(3)	242	No	15(3)	34(3)	No

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) BAW-2082
(2) BAW-1803, Revision 1
(3) BAW-2049
(4) BAW-1920P
(5) Statement not required.

S. E. Yanichko et al, "Analysis of Capsule T from the Rochester Gas and Electric Corporation R. E. Ginna Nuclear Plant Reactor Vessel Radiation Surveillance Program," WCAP-10086, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, April 1982.

S. E. Yanichko et al, "Analysis of Capsule T from the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-10736, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, December 1984.

S. E. Yanichko and V. A. Perone, "Analysis of Capsule V from the Virginia Electric and Power Company Surry Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11415, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, February 1987.

S. E. Yanichko and V. A. Perone, "Analysis of Capsule V from the Virginia Electric and Power Company Surry Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-11499, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, June 1987.

S. E. Yanichko et al, "Analysis of Capsule Y from the Commonwealth Edison Company Zion Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-12396, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, September 1989.

8. CERTIFICATION

This report accurately responds to the request for information stated in Generic Letter 92-01.

6/16/92
M. J. DeVan, Engineer II — Date
Materials and Structural Analysis Unit

6/16/92
A. L. Lowe, Jr., Advisory Engineer — Date
Materials and Structural Analysis Unit

This report was reviewed and found to be accurate.

L B Gross 6/16/92
L. B. Gross, Advisory Engineer — Date
Materials and Structural Analysis Unit

Verification of independent review.

K. E. Moore 6/16/92
K. E. Moore, Manager — Date
Materials and Structural Analysis Unit

This report is approved for release.

D L Howell 6/16/92
D. L. Howell, Project Manager — Date
Owners Group Engineering Services

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1		
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements		
Plant: Zion Unit 2		
Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes □ No ✓
Question II:	Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ✓ No □
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.		
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.		
Question III:	If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response:	Not applicable (see Question I and II above)	
Question IV:	If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response:	Not applicable (see Question I and II above)	

NOTES: WCAP-8132: Surveillance Program Description
(ASTM E 185-70)

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: Zion Unit 2					
Column 1		Column 2	Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
<u>LIMITING BELTLINE WELD</u>					
SA-1769	70 (5)	7, approx.	48	42	An analysis which demonstrates that this material provides margin of safety against fracture equivalent to that required by ASME Section III, Appendix G, has been performed under the sponsorship of the B&W Owners Group's Reactor Vessel Working Group and submitted to the NRC as report BAW-2148P.
<u>LIMITING BELTLINE PLATE OR FORGING</u>					
B8006-1	91 (6)	>32	NA	NA	NA
C4007-1	91 (6)	>32	NA	NA	NA
B8029-1	91 (6)	>32	NA	NA	NA

NOTES FOR TABLE 2 ARE ON THE FOLLOWING PAGE.

TABLE 2 (CONTINUED)

NOTES: (1) Fluence values taken at ¼-thickness.

(2) C_vUSE values for 12/16/91 and EOL (Column 3) calculated per requirements outlined in Regulatory Guide 1.99, Revision 2, Paragraph C.1.2.

(3) C_vUSE is calculated on the basis of RG1.99, Rev. 2, Position 1. SECY 91-333 states that this procedure is "inadequate." Recognizing this and having provided for the uniqueness of Linde 80 welds in BAW-1803, Rev. 1, the method for calculating C_vUSE in BAW-1803, Rev. 1, is put forward as more representative and is intended to be used for predicting the behavior of these welds in licensing applications.

(4) The analysis was based on the worst case weld chemical composition (WF-70) combined with peak fluences seen in the Zion vessels (Units 1 <u>and</u> 2), and is a conservative analysis.

(5) BAW-1803

(6) BAW-10046P

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: Zion Unit 2								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropweight Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
<u>FORGING</u>								
ZV-3855	42,49,32 72,34,32	-7	+10	+25	+10	+10	NB-2331	(1,2,5)
<u>PLATE</u>								
B8006-1	38,36,26 32,38,35	+5	+27	+20	+10	+10	NB-2331	(1,2,5)
B8040-1	36,64,38	-5	+25	+15	-10	-10	NB-2331	(1,2,5)
C4007-1	ND	+30	+68	+65	+10	+10	NB-2331	(1,2,5)
B8029-1	ND	+12	+35	+35	-10	-10	NB-2331	(1,2,5)
<u>WELD</u>								
WF-200	36,35,26	ND	ND	ND	ND	-5	Est. (3)	(1,6)
SA-1769	36,35,38	ND	ND	ND	ND	-5	Est. (3)	(1,7)
WF-154	41,37,43	ND	ND	ND	ND	-5	Est. (3)	(1,6)
WF-70	39,35,44	ND	ND	ND	ND	+18	Eval.(4)	(1,6,8)
WF-29	49,39,45	ND	ND	ND	ND	-5	Est. (3)	(1,6)

NOTES FOR TABLE 3 ARE ON THE FOLLOWING PAGE.

TABLE 3 (CONTINUED)

NOTES TO TABLE 3:

(1) BAW-2150

(2) Mt. Vernon Qualification Test Report

(3) BAW-1803, Revision 1, Tables 3-1 and 3-2; mean of RT_{NDT} values for 34 Linde 80 welds.

(4) BAW-2100

(5) Values are for 60 hr stress-relief.

(6) C_v(+10F) values are for 48 hr stress-relief.

(7) C_v(+10F) values are for 8 - 6 hr cycles stress-relief.

(8) RT_{NDT} value for 40 hr stress-relief maximum.

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: Zion Unit 2		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
ZV3855	1525F-12h/WQ; 1215F-7h/AC; 1125F-29½h/FC	(1,2)
B8006-1	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1100-1150F-31h/FC (cumul.)	
B8040-1	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1100-1150F-31h/FC (cumul.)	
C4007-1	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1100-1150F-29h/FC (cumul.)	
B8029-1	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1100-1150F-29h/FC (cumul.)	
WF-200	1100-1150F-24½h/FC (cumul.)	
SA-1769	1100-1150F-26h/FC (cumul.)	
WF-154	1100-1150F-24½h/FC (cumul.)	
WF-70	1100-1150F-31h/FC (cumul.)	
WF-29	1100-1150F-29h/FC (cumul.)	
SURVEILLANCE MATERIALS		
C4007-1	1600-1650F-9¾h/BQ; 1200-1225F-9¾h/BQ; 1100-1150F-30h/FC	(1)
WF-209-1	1100-1150F-30h/FC	

NOTES:

(1) BAW-2150
(2) Additional stress relief information per Mt. Vernon process drawing.
(3) BQ - brine quench
FC - furnace cool

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: Zion Unit 2

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Lower NB Forging	ZV3855	NB to IS Circ.: WF-200	821T44	8773	(1,2)
IS Plate	B8006-1	IS to LS Circ.: SA-1769	71249	8738	
IS Plate	B8040-1	LS to Dutch Circ.: WF-154	406L44	8720	
LS Plate	B8029-1	IS Longit.: WF-70	72105	8669	
LS Plate	C4007-1	LS Longit.: WF-29	72102	8650	

NOTES: (1) BAW-2150
(2) Mt. Vernon process drawing
(3) NB - Nozzle Belt
IS - Intermediate Shell
LS - Lower Shell

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: Zion Unit 2				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
C4007-1	WF-209-1	72105	8773	(1)

NOTES: (1) BAW-2150

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)										
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>										
Plant: Zion Unit 2										
Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
<u>BELTLINE MATERIAL</u>										
ZV3855	0.22	0.67	0.008	0.006	0.35	0.42	0.66	0.62	0.09	(1)
B8006-1	0.21	1.35	0.010	0.015	0.24	ND	0.54	0.53	0.12	(2)
B8040-1	0.23	1.35	0.008	0.014	0.25	ND	0.52	0.54	0.14	(2)
C4007-1	0.23	1.39	0.010	0.016	0.22	ND	0.53	0.54	0.12	(2)
B8029-1	0.23	1.38	0.010	0.014	0.21	ND	0.51	0.52	0.12	(2)
WF-200	0.07	1.60	0.010	0.015	0.48	0.14	0.63	0.40	0.24	(3)
WF-70	0.09	1.63	0.018	0.009	0.54	0.10	0.59	0.40	0.35	(3)
SA-1769	0.09	1.49	0.020	0.014	0.56	0.16	0.61	0.37	0.26	(3)
WF-29	0.08	1.65	0.015	0.012	0.42	0.05	0.63	0.38	0.23	(3)
WF-154	0.07	1.54	0.013	0.016	0.42	0.07	0.59	0.40	0.31	(3)
<u>SURVEILLANCE MATERIALS</u>										
C4007-1	0.23	1.39	0.010	0.016	0.22	0.065	0.53	0.54	0.12	(4)
WF-209-1	0.08	1.51	0.017	0.013	0.68	0.06	0.57	0.39	0.30	(4)

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). -- Not applicable --

NOTES:

(1) Supplier Material Test Report
(2) BAW-2150
(3) BAW-2121P
(4) BAW-1543, Revision 3

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature

Plant: Zion Unit 2

Cold Leg Temperature (T_{cold}): 529.4 F (See Figure 4-7)

If T_{cold} is <525 F, state how this was considered in determination of embrittlement effects (C_VUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not applicable

References:

None

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: Zion Unit 2
Were surveillance results used in determining C_vUSE? Yes □ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No □
If any "yes" boxes were checked above, state how the surveillance results were used: Initial RT_{NDT} value for WF-70 weld metal
References: BAW-2100

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c							
Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects							
Plant: Zion Unit 2							
Question I. Does measured ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?							
Question II. Does measured C_VUSE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?							
Column 1		Column 2	Column 3	Column 4	Column 5	Column 6	Column 7
Beltline Material	Fluence n/cm^2 (1,2,3)	Measured ΔRT_{NDT}	Predicted $\Delta RT_{NDT}+2\sigma$	Question I If "yes" see Note (4)	Measured C_VUSE Drop	Predicted C_VUSE Drop	Question II If "yes" see Note (4)
ZV-3855	---	ND	ND	--	ND	ND	--
B8006-1	---	ND	ND	--	ND	ND	--
B8040-1	---	ND	ND	--	ND	ND	--
C4007-1	2.57E+18	49(1)	86	No	0(1)	14	No
	8.04E+18	90(1)	110	No	14(1)	19	No
	1.48E+19	121(1)	124	No	0(1)	22	No
B8029-1	---	ND	ND	--	ND	ND	--
WF-200	---	ND	ND	--	ND	ND	--
SA-1769	---	ND	ND	--	ND	ND	--
WF-154	---	ND	ND	--	ND	ND	--
WF-70	6.63E+18	135(2)	259	No	13(2)	22(5)	No
WF-29	---	ND	ND	--	ND	ND	--
Atypical	1.17E+18	28(3)	138	No	9(3)	25(3)	No
	6.56E+18	122(3)	216	No	16(3)	32(3)	No
	7.50E+18	119(3)	223	No	11(3)	32(3)	No
	1.08E+19	120(3)	242	No	15(3)	34(3)	No

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES:

(1) WCAP-12396
(2) BAW-1803, Revision 1
(3) BAW-2049
(4) Statement not required.
(5) BAW-1920P

7. REFERENCES

*A. L. Lowe, Jr. et al, "Analysis of Capsule OC1-F from Duke Power Company Oconee Unit 1 Reactor Vessel Materials Surveillance Program," BAW-1421, Revision 1, Babcock & Wilcox, Nuclear Power Generation Division, Lynchburg, Virginia, September 1975.

*A. L. Lowe, Jr. et al, "Analysis of Capsule ANI-B from Arkansas Power & Light Company's Arkansas Nuclear One, Unit 1 Reactor Vessel Materials Surveillance Program," BAW-1698, Babcock & Wilcox, Nuclear Power Generation Division, Lynchburg, Virginia, November 1981.

*A. S. Heller and A. L. Lowe, Jr., "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds," BAW-1803, Babcock & Wilcox, Utility Power Generation Division, Lynchburg, Virginia, January 1984.

A. L. Lowe, Jr. and J. W. Pegram, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds, BAW-1803, Revision 1, B&W Nuclear Service Company, Lynchburg, Virginia, May 1991.

*J. D. Aadland, "Babcock & Wilcox Owners' Group 177-Fuel Assembly Reactor Vessel and Surveillance Program Materials Information," BAW-1820, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, December 1984.

*A. L. Lowe, Jr. et al, "Analysis of Capsule TE1-A, The Toledo Edison Company, Davis Besse Nuclear Power Station Unit 1, Reactor Vessel Materials Surveillance Program," BAW-1882, Revision 1, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, June 1989.

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*This report is available from B&W Nuclear Service Company, Lynchburg, Virginia.

*A. L. Lowe, Jr. et al, "Analysis of Capsule CR3-C, Florida Power Corporation, Crystal River Unit 3, Reactor Vessel Materials Surveillance Program," BAW-1898, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, March 1986.

*A. L. Lowe, Jr. et al, "Analysis of Capsule CR3-D, Florida Power Corporation, Crystal River Unit 3, Reactor Vessel Materials Surveillance Program," BAW-1899, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, March 1986.

*A. L. Lowe, Jr. et al, "Analysis of Capsule TMI1-C, GPU Nuclear, Three Mile Island Nuclear Station-Unit 1, Reactor Vessel Materials Surveillance Program," BAW-1901, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, March 1986.

*A. L. Lowe, Jr., "Reactor Pressure Vessel and Surveillance Program Materials Information for Surry Units 1 and 2, North Anna Units 1 and 2," BAW-1908, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, February 1986.

*A. L. Lowe, Jr., "Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for Surry Units 1 and 2," BAW-1909, Revision 1, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, August 1986.

*A. L. Lowe, Jr. et al, "Analysis of Capsule CR3-LG1, Babcock & Wilcox Owners Group, Integrated Reactor Vessel Materials Surveillance Program," BAW-1910P, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, August 1986.

*A. L. Lowe, Jr. et al, "Analysis of Capsule DB1-LG1, Babcock & Wilcox Owners Group, Integrated Reactor Vessel Materials Surveillance Program," BAW-1920P, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, October 1986.

*A. L. Lowe, Jr. et al, "Analysis of Capsule CR3-F, Florida Power Corporation, Crystal River Unit 3, Reactor Vessel Materials Surveillance Program," BAW-2049, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, September 1988.

*A. L. Lowe, Jr. et al, "Analysis of Capsule OC1-C, Duke Power Company, Oconee Nuclear Station Unit 1, Reactor Vessel Materials Surveillance Program," BAW-2050, Babcock & Wilcox, Nuclear Power Division, Lynchburg, Virginia, October 1988.

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