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April 23, 1986

Mr. John F. Stolz, Director
PWP Project Directorate #6
Division of PWR Licensing - B
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Reactor Vessel Closure Region Thermal Stress during Natural Circulation Cooldown - Generic Issue #79

Reference: (1) Letter, J. F. Stolz to C. H. Turk, same subject, dated March 27, 1986

Dear Mr. Stolz:

The information requested in Reference 1 is being provided by the B&W Owners Group Analysis Committee to assist in your review of the subject generic issue.

The materials properties requested are contained in reports submitted to the NRC by the B&W Owners Group Materials Committee, namely BAW-1820 and BAW-10046, Rev. 2, a copy of which is attached. Adjusted RT_{NDT} vary from plant to plant and is available in submitted Reactor Vessel Surveillance Capsule reports available through your B. J. Elliot and P. N. Randall.

The geometric parameters of reactor vessel components requested via References 7 through 15, 19, 27, and 32 of B&W calculational file 32-1151155-00 are also provided to assist you in building your model.

Aperture Card Dist

G006 Drawings
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This information is provided solely at your request and is for your information in addressing Generic Issue #79.

Very truly yours,

R J Schomaker

R. J. Schomaker
Project Manager
Owners Group Engineering Services

RJS/leh

Attachment

cc: B&WOG Analysis Committee

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BAW-10046, Rev 2

Topical Report
December 1984

Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G

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BAW-10046, Rev 2

Topical Report
December 1984

METHODS OF COMPLIANCE WITH FRACTURE TOUGHNESS AND OPERATIONAL REQUIREMENTS OF 10 CFR 50, APPENDIX G

by

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Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G

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Key Words: Ferritic Materials, Reactor Coolant Pressure Boundary, Reference Temperature, Charpy Upper Shelf Energy, Appendix G to 10 CFR 50, Appendix G to ASME Code, Fracture Prevention, Pressure Temperature Limitation, Technical Specifications, Ductile Tearing Instability, Elastic-Plastic Fracture Mechanics, Deformation Plasticity Failure Assessment Diagram

ABSTRACT

This report describes B&W's practices, methods, and criteria for compliance with the requirements of Appendix G to 10 CFR 50, "Fracture Toughness Requirements." The ferritic materials and the operational parameters of the reactor coolant system for nuclear power plants designed by B&W are described as are the methods for obtaining and estimating the reference temperature and the Charpy upper shelf energy. The acceptance criteria for unirradiated Charpy upper shelf energy is given. The adequacy of fracture toughness properties of bolting materials and type 403 materials are demonstrated. The methods employed to determine the reactor coolant system pressure-temperature limit curves are given for each of the loading conditions required by Appendix G to 10 CFR 50. The pressure-temperature limit curves imposed by several regions of the reactor vessel are illustrated as is the development of the composite limit curves. Furthermore this report describes the methods used to preclude ductile tearing instability. This analysis applies to irradiated vessels with low upper shelf energies. The Technical Specifications pressure-temperature limit curves and the Preservice System Hydrostatic Test limit curve of a typical 177 FA plant are also described.

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1. INTRODUCTION

1.1. Background

On July 17, 1973, a new appendix to 10 CFR 50, entitled "Appendix G - Fracture Toughness Requirements" was published in the Federal Register. 10 CFR 50, Appendix G has been revised in subsequent years. This report reflects the revised criteria including effective issue July 26, 1983. This appendix specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of water-cooled power reactors and provides specific guidelines for determining pressure-temperature operational limitations on the RCPB. The toughness and operational requirements are specified to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the RCPB may be subjected over its service lifetime. Although the requirements of Appendix G became effective August 13, 1973, they are applicable to all boiling water and pressurized water-cooled nuclear power reactors, including those under construction or in operation on the effective date.

At the time 10 CFR 50, Appendix G, became effective, immediate compliance with some of its provisions was not possible for plants whose pressure boundary components were ordered in accordance with an edition or addenda of Section III of the ASME Boiler and Pressure Vessel Code (hereafter ASME Code) published before the Summer 1972 Addenda. For these plants, neither the fracture toughness data required by Appendix G nor the material for performing toughness tests is available. Also, the stress calculations required to quantitatively define the allowable pressure at any given temperature were not readily available. Appropriate, conservative methods of compliance for these plants have been developed and are described in this report.

1.2. Scope and Organization

This report presents B&W's practices, methods, and criteria for compliance with the requirements of 10 CFR 50, Appendix G. It is applicable to all current B&W nuclear steam systems (NSSS). The definitions and terminology of 10 CFR 50, Appendix G, and the ASME Code are used whenever appropriate.

The report is divided into seven parts and is summarized in Part 7. Part 2 describes the reactor coolant pressure boundary (RCPB) and includes a list of the components and ferritic materials used in their construction. Part 2 also describes the operational modes of the RCPB related to nonductile failure for each of the loading conditions for which pressure-temperature limit curves are required.

Part 3 presents the fracture toughness properties of the ferritic materials of the RCPB. These materials are grouped as follows:

1. Ferritic materials other than (a) bolting and (b) type 403 stainless steels
2. Bolting materials
3. Type 403 stainless steel

For the first group, Part 3 describes methods for (1) determining the unirradiated reference temperature (RT_{NDT}) for the ferritic materials and the unirradiated Charpy upper shelf energy (C_vUSE) level of the beltline region materials. The justification for use and acceptance criteria for unirradiated beltline region materials with C_vUSE lower than 75 ft-lbs are presented.

For the second group, bolting materials, Part 3 presents justification for allowing the lowest service temperature, and the minimum preload temperature to be 40F. The impact properties of these materials are also presented.

For the third group of materials, Part 3 includes a demonstration of adequate fracture toughness properties.

Part 4 presents the basis for a step-by-step description of the calculational procedure to determine the pressure-temperature limitations of the reactor coolant system; this is done to ensure adequate fracture toughness under the loading conditions of interest.

Part 5 gives an example of beginning- and end-of-life pressure-temperature limit curves that were developed using the material properties in Part 3 and the calculational procedure of Part 4. Similar curves were developed for each plant and conservatively adjusted for use in the Technical Specifications issued by the Nuclear Regulatory Commission (NRC) as a part of the plant operating license. Typical limit curves, as they appear in the Technical Specifications, and the limit curve for the preservice system hydrostatic test are shown in Part 5.

Part 6 presents the supplemental analysis performed in the event a reactor vessel beltline is predicted to be below 50 ft-lbs upper shelf. This analysis is an elastic plastic fracture mechanics assessment confirming that the vessel has sufficient toughness to preclude ductile tearing instability.

2. REACTOR COOLANT PRESSURE BOUNDARY

2.1. Components

The RCPB is defined by NRC Regulation 10 CFR 50.2, (v) as follows:

> "'Reactor coolant pressure boundary' means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessel, piping, pumps, and valves, which are:
>
> (1) Part of the reactor coolant system, or
>
> (2) Connected to the reactor coolant system, up to and including any and all of the following:
>
> (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,
>
> (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
>
> (iii) The reactor coolant system safety and relief valves.
>
> For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping."

The reactor coolant system (RC system) for B&W nuclear power plants is made up of the following components: reactor vessel, steam generators, pressurizer, reactor coolant pumps, valves and interconnecting piping. The RC system contains and circulates reactor coolant at the pressure and velocity necessary to transfer the heat generated in the reactor core to the secondary fluid in the steam generators.

The other pressure-containing portions of the RCPB are the auxiliary system components. These include the makeup and purification system piping and valves (including RC pump seal injection lines); the emergency core cooling system high- and low-pressure and core flooding injection piping and core flooding injection piping and valves; the vent, drain, and other piping and valves used for maintaining the RC system; and the incore instrumentation on piping.

Portions of the RCPB are exempted from the requirements for Class 1 components of ASME Code Section III by footnote 2 to NRC Regulation 10 CFR 50.55a, which reads as follows:

Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary defined in 50.2(v) need not meet these requirements, provided:

(a) In the event of postulate failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only, or

(b) the component is or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

Components of the RCPB included under this exemption provision are generally designed and fabricated in accordance with the requirements for Class 2 components in ASME Code Section III (see Regulatory Guide 1.29, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste containing Components of Nuclear Power Plants"). None of these components are constructed of ferritic material except in some instances the core flood tanks, which are carbon steel in some B&W plants. Although the core flood tanks are isolated from the RC system by two valves during normal operation, connecting piping to the tanks (1-inch lines for nitrogen addition fill and drain) does penetrate reactor containment. Therefore, the system is part of the RCPB to the outermost containment isolation valve. Since these tanks are isolated from the RC system during all conditions of normal operation, including anticipated operational occurrences, they need not be considered in developing the RC system pressure-temperature limitations and are not discussed in this report.

2.2. Ferritic Materials and RCPB Operational Parameters

The ferritic materials used in construction of the RCPB for B&W nuclear power plants are listed for each component in Table 2-1. The pressure boundary of the RC system is fabricated primarily from ferritic materials, while that of the auxiliary systems is fabricated primarily from austenitic material.

Consequently, the RC system components are the only ones that require special protection against nonductile failure and that must comply with the fracture toughness requirements of 10 CFR 50, Appendix G. This protection against nonductile failure is ensured by imposing pressure-temperature limitations on operation of the RC system. The margin of safety is controlled by the maximum calculated allowable pressure at any given temperature. The following loading conditions require pressure-temperature limits:

1. Normal operations including bolt preloading, heatup and cooldown.
2. Preservice system hydrostatic test.
3. Inservice system leak and hydrostatic tests.
4. Reactor core operation.

To impart a better understanding of the required protection against nonductile failure, typical operational parameters of the RC system are described in the following paragraphs for each of the loading conditions.

2.3. Normal Operation

2.3.1. Bolt Preload

During bolt preload, the reactor vessel closure studs are tensioned to the specified load. Bolt preloading is not allowed until the reactor coolant temperature and the volumetric average temperature of the closure head region (including the studs) is higher than the specified minimum preload temperature. After the studs are tensioned, system pressure can be increased by the pressurizer until it is above the net positive suction head (NPSH) required for RC pump operation. The heatup transient begins when the RC pumps are started.

2.3.3. Heatup

During heatup the RC system is brought from a cold shutdown condition to a hot shutdown condition. The heat sources used to increase the temperature of the system are the RC pumps and any residual (decay) heat from the core. Normally, when the pumps are started, the temperature of the water in the pressurizer is about 400F; this corresponds to the pressure in the RC system of about 300 psig. The coolant temperature is at or above the minimum specified bolt preload temperature. Initially, the reactor coolant temperature may be as low as room temperature for initial core loading or as high as 130F for subsequent refueling.

At any given time throughout the heatup transient, the temperature of the reactor coolant is essentially the same throughout the system except, of course, in the pressurizer. The system pressure, as controlled by the pressurizer heaters is maintained between the minimum required for RC pump NPSH and the maximum to meet the fracture toughness requirements. The heatup rate is maintained below the maximum rate used to establish the maximum allowable pressure-temperature limit curve.

2.3.3. Cooldown

RC system cooldown brings the system from a hot to a cold shutdown condition. The cooldown is normally accomplished in two phases. The first phase reduces the fluid temperature from approximately 550F to below the design temperature of the decay heat removal system (approximately 300F). This temperature reduction is accomplished using the steam generators but bypassing the turbine and dumping the steam directly to the condenser. Once below its design temperature (and pressure), the decay heat removal system (DHRS) is activated in the second phase to further reduce the reactor coolant temperature to that desired.

Before cooldown, the RC system temperature is maintained constant by balancing the heat removal rate from the steam dump with the heat contributed by the RC pumps and core decay heat. The system pressure is maintained by the pressurizer. The cooldown is normally initiated by stopping one RC pump in each loop. The two remaining pumps provide coolant circulation through both steam generators, and the turbine steam bypass flow controls the cooldown rate. The primary pressure during cooldown is controlled with the pressurizer heaters and spray. After cooling down below the DHRS design temperature and pressure, the cooling mode is changed from the steam generators to the DHRS. Before the switch, the RC system pressure is below 625 psig (20% of the preoperational system hydrostatic test pressure) and below the DHRS pressure but above the pressure required for the RC pumps to operate.

To minimize the thermal shock on the RCPB, the two RC pumps remain in operation as the water flow of the DHRS is initiated. The DHRS flow rapidly mixes with the reactor coolant, but during this period, the indicated RC temperature may fluctuate until mixing is complete. After the switch is completed, the RC pumps are stopped. During this phase, the cooldown rate is controlled by the temperature and flow of the DHRS.

2.4. Preservice System Hydrostatic Test

Prior to initial operation, the RC system is hydrostatically tested in accordance with ASME Code requirements. During this test, the system is brought up to an internal pressure not less than 1.25 times the system design pressure. This minimum test pressure is in accordance with Article NB-6000 of ASME Section III. Since the system design pressure is 2500 psig, the preservice system hydrostatic test pressure is 3125 psig. Initially, the RC system is heated to a temperature above the calculated minimum test temperature required for adequate fracture toughness. This heatup is accomplished by running the RC pumps. The pressurizer heaters are used to heat the pressurizer to the required temperature. Before the test temperature is reached, the pressure is maintained above the NPSH required for the RC pumps but below the maximum allowable pressure for adequate fracture toughness. When the test temperature is reached, the RC pumps are stopped and RC makeup water is added to fill the pressurizer. The test pressure is then reached using either the pressurizer heaters or the hydrostatic pumps connected to the RC system. The test pressure is held for the minimum specified time, and the examination for leakage follows in accordance with the ASME Code.

2.5. Inservice System Leakage and Hydrostatic Tests

When inservice system leakage tests are required, the system is brought from a cold to a hot shutdown condition. The means of heating the system and increasing the pressure are the same as those used during normal heat-up. If it is necessary to cool the system down after either test, normal cooldown procedures are used. These tests are conducted in accordance with the requirements of ASME Section XI, Article IWA-5000. The test pressure for the inservice leakage tests is the pressure that, for the component located at the highest elevation in the system, is no less than the system nominal operating pressure at 100% rated reactor power. For the inservice hydrostatic test, ASME Section XI gives a table (Table IWB-5222-1) of the minimum test pressure versus the test temperature at which the system must be tested. The test temperature for both the inservice leakage and hydrostatic tests is determined by the requirements for fracture toughness.

2.6. Reactor Core Operation

The reactor core is not allowed to become critical until the RC system fluid temperature is above 525F except for brief periods of low-power physics testing. This temperature is much higher than the minimum permissible temperature for the inservice system hydrostatic pressure test, and it is also at least 40F above the calculated minimum temperature required at normal pressure for operation throughout the service life of the plant.

Table 2-1. Ferritic Materials Used in Reactor Coolant Pressure Boundary

Component	Material
Reactor Vessel	
Plates	SA 533, Grade B, Class 1
Forgings	SA 508, Class 2; SA 182, Grade F6
Bolting	SA 540, Grade B-23 or -24
Welds	SFA 5.5, SFA 5.17
Bars	A276 Type 403 (Code Case 1337 or N-4)
Steam Generator	
Plates	SA 533, Grade B, Class 1; SA 516, Grade 70
Forgings	SA 508, Class 1
Bolting	SA 540, Grade B-23 or -24
Welds	SFA 5.5, SFA 5.17
Pressurizer	
Plates	SA 533, Grade B, Class 1
Forgings	SA 508, Class 2
Bolting	SA 540, Grade B-23; SA 320, Grade L43
Welds	SFA 5.5, SFA 5.17
Reactor Coolant Piping	
Plates	SA 516, Grade 70
Forgings	SA 105, Grade 2
Seamless Pipe & Tubing	SA 106, Grade C
Welds	SFA 5.5, SFA 5.17
Reactor Coolant Pump	
Forgings	SA 508, Class 2; SA 350, Grade LF2
Bolting	SA 540, Grades B-21, -23, -24
Valves	
Forgings	SA 105 Grade 2

3. MATERIAL PROPERTIES

3.1. Impact Properties of Ferritic Materials

To determine the pressure-temperature operating limitations for the RCPB the reference nil-ductility temperature (RT_{NDT}) of the ferritic materials must be established. The RT_{NDT} is needed to calculate the critical stress intensity factor (K_{IR}). In ASME Appendix G, K_{IR} is related to temperature, T, and to RT_{NDT} by the following equation:

$$K_{IR} = 26.77 + 1.223 \exp[0.0145(T - RT_{NDT} + 160)] \text{ksi}\sqrt{\text{in}}.$$

This relationship is applicable only to ferritic materials that have a specified minimum yield strength of 50,000 psi or less at room temperature.

Since the impact properties of the beltline region materials of a reactor vessel will change throughout its lifetime, periodic adjustments are required on the pressure-temperature limit curves of the RCPB. The magnitude of these adjustments is proportional to the shift in RT_{NDT} caused by neutron fluence. Therefore, it is essential to determine the radiation-induced ΔRT_{NDT} of the beltline region materials.

Since the ΔRT_{NDT} is based on the temperature shift of the Charpy curves measured at the 30 ft-lb level, it is necessary to know, by analysis or from the results of the material surveillance program, the magnitude of the Charpy 30 ft-lb shift.

3.1.1. Determination of RT_{NDT}

3.1.1.1. ASME Code Method

The RT_{NDT}s of the ferritic materials, which were specified and tested in accordance with the fracture toughness requirements of the ASME Section III Summer 1972 Addenda (to 1971 Edition) or later Editions and Addenda, are determined as required by that Code. When sufficient material is available, the RT_{NDT}s of the beltline region materials (which were specified and tested in accordance with an Edition or Addenda of ASME Section III earlier than the Summer 1972 Addenda) are obtained by testing specimens oriented normal to the principal working direction. The test procedure is in accordance with ASME Section III, paragraph NB 2300 (Summer 1972 or later Edition and Addenda).

3.1.1.2. Estimating Method

The RCPBs of several plants were designed and constructed in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda. Except for the beltline region materials for which sufficient test material is available, the RT_{NDT}s of the ferritic materials must be estimated. This is necessary because obtaining the test data required for the exact determination of RT_{NDT} was not required by the applicable ASME Code. Generally, drop weight tests were not performed, and the Charpy V-notch tests were limited to "fixed" energy level requirements for specimens oriented in the longitudinal (principal working) direction at a temperature of 40F or lower.

To obtain an RT_{NDT} estimate that is appropriately conservative, B&W has collected and evaluated the data from tests conducted on pressure-retaining ferritic materials to which the new fracture toughness requirements were applied.

3.1.1.3. Estimated RT_{NDT}

In the preceding section pertinent impact data for each type of ferritic material are discussed as a basis for estimating conservative RT_{NDT}s. Estimated RT_{NDT}s are needed for all materials that were specified to meet the requirements of an Edition or Addenda of ASME Section III earlier than the Summer 1972 Addenda. This section summarizes the data and the estimated RT_{NDT} of the ferritic materials used in construction of the RCPB.

The data are summarized in Table 3-1. For each type of material, the table lists the number of cases considered; the highest measured RT_{NDT}; the average of the measured RT_{NDT}s; the estimated RT_{NDT}; and the difference between the average measured and the estimated temperatures.

3.1.2. Determination of Charpy V-Notch Level

3.1.2.1. Specified Method

Appendix G to 10 CFR 50 requires complete characterization of the unirradiated impact properties of all the beltline region materials of the reactor vessel. This includes determination of RT_{NDT} and Charpy (C_v) test curves for the directions normal to and parallel to the principal working direction (other than the thickness direction). Appendix G also requires a minimum Charpy upper shelf energy (C_vUSE) of 75 ft-lb for all beltline region materials unless it is demonstrated that lower values of upper shelf fracture energy provide an adequate margin against irradiation induced degradation.

To comply with Appendix G, the beltline region materials (not including HAZ) of reactor vessels for later plants meet the following test requirements:

> In addition to the Charpy V-notch impact tests needed to determine RT_{NDT}, 15 Charpy V-notch impact tests shall be conducted in each required direction (for base metals the required directions are normal and parallel to the principal direction in which the material was worked, other than the thickness direction). The tests shall be conducted at appropriate temperatures over a temperature range sufficient to define the C_v test curves (including upper shelf levels) in terms of both fracture energy and lateral expansion. Three specimens shall be tested at each test temperature for the determination of RT_{NDT}. The Charpy upper shelf energy shall be determined as follows:
>
> (1) Two sets of three Charpy specimens each shall be tested at two temperatures at which the percent of shear fracture is approximately 95%. The Charpy upper shelf energy shall be the higher average energy value of the two sets of Charpy specimens.
>
> (2) If either of the two average upper shelf energy values of step (1) is below 75 ft-lb, another set of three Charpy specimens shall be tested at a temperature at least 50F higher than the highest temperature of step (1). The Charpy upper shelf energy shall be the highest average value of the three sets of Charpy specimens.
>
> The location and orientation of the impact test specimens shall comply with the requirements of paragraph NB-2322 of Section III of the ASME Code.

The requirements for the minimum C_vUSE are described in section 3.1.3.3. The requirements above are also met for the HAZ of the beltline region base metal(s) that are selected to be monitored by the reactor vessel surveillance program. The requirements are not specified for the HAZ of the other beltline region materials because the ASME Code (Paragraph NB-4335 of the Winter 1974 Addenda) deleted the requirements for toughness testing of HAZs in the weld procedure qualification tests. B&W has elected to follow the new ASME requirements.

For the beltline region materials of reactor vessels that were specified in accordance with the requirements of an Edition or Addenda of ASME Section III issued before the Summer 1972 Addenda, the complete C_v test curves, including C_vUSE, is determined when the material is included in the reactor vessel material surveillance program. For the beltline region materials that are not included in the surveillance program, and when sufficient material is available, the C_v test curve and USE are determined only in the direction normal to the principal working direction. No minimum C_vUSE is required, other than the 50 ft-lbs/35 mils of lateral expansion for the beltline region materials of these reactor vessels, one of the conditions required to establish RT_{NDT}. When the unirradiated C_vUSE of these materials is below 75 ft-lb, the currently accepted procedure is applied to predict the end-of-service C_vUSE.

3.1.2.2. Estimating Method

The C_v USE must be estimated for reactor vessel beltline region materials that were specified in accordance with the requirements of an Edition or Addenda of ASME Section III issued before the Summer 1972 Addenda and for which insufficient material is available for testing. All available data from tests conducted on reactor vessel beltline region materials were collected and evaluated in order to obtain an appropriately conservative estimate. Not all the data were obtained in accordance with the methods specified in section 3.1.2.1 since in some cases the absorbed energy was obtained only at one temperature.

3.1.2.3. Estimated C_vUSE

The data used for estimating conservative C_vUSE is discussed in the preceding section. The estimated C_vUSE is needed for all of those beltline region materials for which test material is not available, i.e., for which the actual C_vUSE data and the estimated energy for each type of beltline region material are summarized in Table 3-2. For each type of material, the table lists the number of tested heats, the lowest measured, average measured, and estimated C_vUSEs and the average difference between the estimated and measured C_vUSE.

3.1.3. Radiation Effects

3.1.3.1. Adjustment of RT_{NDT}

Adjustment of the RT_{NDT} to accommodate the radiation-induced changes in fracture toughness of beltline region materials is an important factor in developing pressure-temperature limits. Correlations have been developed for predicting the radiation-induced RT_{NDT} to be used in adjusting the initial RT_{NDT} for pressure-temperature analyses. These correlations are not perfected and, therefore, subject to continuous updating as additional data and information is developed.

The methodology used to adjust the RT_{NDT} values as used in developing pressure-temperature limits will be the currently accepted procedures. The method used will be referenced in all pressure-temperature analyses and will be reported in the Owners licensing documents.

3.1.3.2. Decrease in C_vUSE

Neutron irradiation of the beltline region materials cause a decrease in C_vUSE. Correlations have been developed for predicting this decrease in Charpy USE. These correlations are not perfect and, therefore, are subject to updating as additional data and information is obtained.

The methodology used to predict the decrease in C_vUSE (used in the evaluation of beltline region materials) will be the currently accepted procedures. The method used will be referenced in all analyses and will be reported in the Owners licensing documents.

3.1.3.3. Acceptance Criterion for Unirradiated C_vUSE

Appendix G to 10 CFR 50 requires that the C_vUSE of the unirradiated beltline region materials be equal to or greater than 75 ft-lb except if it is demonstrated by appropriate data and analyses that lower values still provide adequate margin for degradation resulting from neutron irradiation.

This section demonstrates that for some beltline region materials, a C_vUSE lower than 75 ft-lb still provides an adequate margin for degradation from irradiation. This section also presents an acceptance criterion for C_vUSE lower than 75 ft-lb which is applied to later plants.

The beltline region of the reactor vessel includes all the ferritic material in the reactor vessel that (1) directly surrounds the effective height of the active core and (2) adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in selecting the limiting material with regard to radiation damage. The beltline region material above and below the effective height of the fuel element assemblies are irradiated to a neutron fluence received by the material directly surrounding the fuel element assemblies. Since not all beltline region material is subjected to the same neutron fluence, it is not necessary for all of this material to have a C_vUSE greater than 75 ft-lb. Also, the radiation-induced drop in C_vUSE depends not only on the neutron fluence but on the material's chemical composition. The required C_vUSE of unirradiated beltline region materials is defined in terms of the material's chemical composition and the predicted end-of-service neutron fluence to which the material will be subjected.

Complete Charpy V-notch impact curves are required for all of the unirradiated beltline region materials used in later reactor vessels. The test requirements are in accordance with Appendix G to 10 CFR 50 and are described in section 3.1.2.1. C_vUSE requirements are as follows:

1. The C_vUSE of the beltline region materials directly surrounding the effective height of the fuel assemblies shall be equal to or greater than 75 ft-lb.

2. The C_vUSE of the beltline region materials above and below the effective height of the fuel assemblies shall be equal to or greater than the sum of the following energies:

   a. The energy calculated using the material's chemical composition, end-of-service neutron fluence at the 1/4T vessel wall location, and an accepted prediction technique which will provide an end of service life C_vUSE no less than 50 ft-lbs.

   b. The energy equivalent to 5% of the energy calculated in step a.

The minimum C_vUSE above provides adequate margin for degradation from irradiation. All the beltline region materials of later reactor vessels have been specified to have a low copper content ($\leq$0.10%), and the predicted drop in C_vUSE is very small for the neutron fluence of interest.

3.2. Impact Properties of Bolting Materials

3.2.1. Code Requirements

Appendix G to 10 CFR 50 requires that materials for bolting and other fasteners meet the ASME Code. In the early editions of the ASME Code, up to and including the Winter 1971 Edition, it was required that the bolting materials exhibit a "fixed" minimum average energy at a temperature of 10F. One specimen in a set of three was allowed to be less than the fixed ft-lb value, but not less than the fixed value minus 5 ft-lb. In the Summer 1972 Addenda to the 1971 Edition, the fracture toughness requirements for bolting materials were changed to be consistent with the requirements of Appendix G except that no requirements were made in terms of absorbed energy (ft-lb). The requirements were changed again by the Summer 1973 Addenda to the 1971 Edition. In this revision and subsequent editions of ASME Section III, 45 ft-lb absorbed energy was required only for bolting materials having a nominal diameter greater than 4 inches.

All bolting materials ordered after the effective data of Appendix G to 10 CFP 50 (August 16, 1973) meet the requirements of Appendix G. Bolting materials ordered before this date must meet the requirements of the applicable ASME Code.

3.2.2. Estimating Method

To establish the minimum preload temperature and the lowest service temperature of a pressure-retaining component, it is necessary to know the lowest temperature at which the bolting materials have adequate fracture toughness. This lowest temperature is either the temperature at which the bolting materials exhibit a 25-mil lateral expansion and 45 ft-lb absorbed energy or the temperature at which the bolting materials are at the C_vUSE. For bolting materials of pressure-retaining components ordered before August 16, 1973, it is necessary to estimate the lowest temperature at which these Charpy impact properties are met. The preload temperature and the lowest service temperature are defined by the applicable equipment specification for components ordered after August 16, 1973.

Impact data from 13 heats of SA 540 Class 3 bolting were evaluated in order to estimate the lowest temperature at which bolting materials have adequate fracture toughness. The principal criteria defining the fracture toughness requirements for the bolting materials used in the reactor coolant pressure boundary are described in WRC Bulletin 175.[3] The fracture mechanics analysis performed and described in WRC Bulletin 175 shows that for the reference flaw size of 0.3 inch (nominal diameter over 3 inches), the required fracture toughness (K_{IC}) is about 125 ksi$\sqrt{in.}$ for bolting materials with a specified minimum yield strength of 130 ksi. To protect against nonductile failure, fracture toughness values exceeding 125 ksi$\sqrt{in.}$ would be needed at the lower service temperature at which maximum Code-allowed stresses occur. In WRC Bulletin 175 K_{IC} versus C_v energy correlations were used to estimate the C_v energy that would correspond to 125 ksi$\sqrt{in.}$ The K_{IC} versus C_v correlations were those of Barson and Rolfe.[4] Their empirical correlations are between slow-bend K_{IC} tests and the results of standard Charpy V-notch impact tests for the transition-temperature and upper shelf regions. The transition-temperature K_{IC}-CVN correlation is

$$\frac{(K_{IC})^2}{E} = 2(CVN)^{3/2} \tag{1}$$

and the upper shelf K_{IC}-CVN correlation is

$$\left(\frac{K_{IC}}{\sigma_y}\right)^2 = \frac{5}{\sigma_y}\left(CVN - \frac{\sigma_y}{20}\right) \tag{2}$$

The relationship in equation 1 suggests that at the transition-temperature region of the Charpy curve, 41 ft-lb corresponds to 125 ksi$\sqrt{in.}$ For the upper shelf region of the Charpy curve, the relationship of equation 2 relates 28 and 30 ft-lb to 125 ksi$\sqrt{in.}$ for bolting materials having yield strengths of 160 and 130 ksi, respectively.

Even though two of the bolting material heats evaluated do not meet the requirements of Appendix G, the materials have adequate fracture toughness to provide a conservative margin of safety against nonductile failure. At +40F, the bolting materials evaluated are at the upper shelf region of their C_v test curves. For the bolting materials under consideration, C_vUSEs of 28 ft-lb would have sufficient fracture energy to prevent failure because the upper shelf K_{IC}-CVN correlation shows that 28 ft-lb corresponds to 125 ksi$\sqrt{in}$. The lowest C_vUSE of the data collected, 42 ft-lb, corresponds to a fracture toughness value of 165 ksi$\sqrt{in}$. To ensure adequate margin of safety, the lowest service temperature and the minimum preload temperature are defined to be higher than 40F.

3.3. Impact Properties of Type 403 Modified Steel

3.3.1. Code Requirements

Appendix G to 10 CFR 50 requests that the adequacy of the fracture toughness properties of ferritic materials such as Type 403 modified stainless steel be demonstrated to the Commission on a case-by-case basis. The Type 403 modified steel is used as a RCPB material in the motor tube of the control rod drive mechanism. This section demonstrates that, for this application, the material has adequate fracture toughness for protection against non-ductile failure.

The nominal wall thickness of the motor tube section of interest is more than 1/2 inch and less than 5/8 inch. In the early editions of ASME Section III up to the Winter 1971 Addenda to the 1971 Edition, materials with a nominal section thickness of 1/2 inch or less did not require impact testing. Starting with the Summer 1972 Addenda, the nominal section thickness increased to 5/8 inch or less. Thus, in the early editions of ASME Section III, the Type 403 modified steel required impact testing, but in the new editions it does not. However, since this material was selected for use, B&W has ordered it to meet the impact toughness requirements of ASME Section III, as if its nominal wall thickness exceeded 5/8 inch. For materials order to ASME Section III, Summer 1972 and later Addenda, the imposed acceptance standard for nominal wall thicknesses from 5/8 to 3/4 inch, inclusive, is presented in Paragraph NB-2332. The material has also been specified to meet the requirements of SA 182 Grade F6 (forgings) or ASTM A276 (bars) as modified by ASME Code Case 1337.

When ordered according to the early revisi: of Code Case 1337 (including Revision 6) and to the early editions of ASME Section III, the Type 403 modified forgings or bars were required to be impact-tested at 20F. The minimum average energy of a set of three Charpy V-notch specimens was 35 ft-lb, with one specimen allowed to be less than 35 but not less than 30 ft-lb. For both forgings and bars, the Charpy specimens were oriented in the axial (longitudinal) direction.

In the Summer 1972 Addenda to the 1971 Edition of ASME Section III, the fracture toughness requirements of all pressure boundary ferritic materials changed; however, no acceptance criterion was given for the martensitic high-alloy chromium steels, such as Type 403 modified steel. A year later, the Summer 1973 Addenda re-established the acceptance criteria for the type 4XX steels. Beginning with this addenda, the fracture toughness requirements and acceptance criteria for the type 4XX steels are described in Paragraph NB-2332 of ASME Section III. This paragraph requires that three Charpy V-notch specimens be tested at temperatures lower than or equal to the lowest service temperature. The lateral expansion of each specimen must be equal to or greater than 20 mils. The test temperature has been specified as equal to or less than 40F. The orientations of the specimens are transverse (normal to principal working direction) for the forgings and axial for the steel bars. The fracture toughness requirements of Code Case Summer 1337, starting with Revision 7, are the same as those of ASME Section III, Summer 1973 Addenda to the 1971 Edition.

3.3.2. Demonstration of Adequate Toughness

It is B&W's position that the fracture toughness requirements of the new editions of ASME Section III provide adequate protection against nonductile failure. The proof of adequate toughness is based on demonstrating that the Type 403 modified steels used in the construction of components designed to an Edition or Addenda of ASME Section III prior to the Summer 1973 Addenda meet or exceed the toughness requirements of that Addenda.

Data from 15 lots of SA 186 F6 forgings and 15 lots of ASTM A276 bars were evaluated. Based on these data, the lowest service temperature of the control rod drive mechanism can be as low at 40F; however, for additional protection against non-ductile failure, B&W has defined the component's lowest service temperature at 100F. This specified lowest service temperature is

60F above the temperature at which the fracture toughness requirements are specified and met. The additional 60F provides margins of safety beyond that required by the ASME Code and by Appendix G to 10 CFR 50.

3.4. Supplemental Fracture Toughness Properties

In the event the beltline material reaches a radiation level which causes the predicted Charpy upper shelf energy value to decrease below 50 ft-lb at 1/4T, supplemental fracture toughness data will be obtained to assess reactor pressure vessel integrity. The data are used to demonstrate equivalent margins of safety as established in Appendix G of ASME Code.

3.4.1. Terminology Related to Ductile Fracture Analysis

The terminology used in the development of material properties for analysis of the reactor vessel resistance to ductile fracture will be in accordance with the following standards.

3.4.1.1 Mechanical Properties -- ASTM Specification E6, Standard Definitions of Terms Relating to Methods of Mechanical Testing

3.4.1.2 Fracture Toughness Properties -- ASTM Specification E616, Standard Terminology Relating to Fracture Testing

3.4.2. Fracture Toughness Properties of Ductile Materials

When the beltline region materials of the reactor pressure vessel reach an irradiation level which causes the Charpy upper shelf energy value of the material to decrease to a value below 50 ft-lb, supplemental fracture toughness data to assess reactor vessel integrity are required by 10 CFR 50. These data are used to provide input to the elastic-plastic fracture mechanics analysis as described in section 6.

The data base for this fracture mechanics analysis is being developed in the integrated reactor vessel material surveillance program described in BAW-1543 and the interpretation of the materials data obtained from this surveillance program will be presented in RVSP reports. The data that are most important to the analysis are those which define the initiation of ductile tearing and the resistance of the material to ductile tearing as a function of crack growth. The interpretation of the data is presented in the load- displacement curves obtained from the individual tests and the resulting J-R curves derived from the data. Supporting data is obtained from the stress-strain curves of the tension tests. These data are analyzed to obtain the true stress-true strain curves to provide the work hardening coefficients.

The actual material properties used in the establishment of the reactor pressure vessel operating limitations and supporting references will be reported in the appropriate licensing document.

3.4.3. Relationship Between Fracture Toughness Properties and the Fracture Mechanics Analysis

The technical approach used in Appendix G of 10 CFR 50 is to establish the reactor vessel operating limitations with adequate margins of safety using a fracture mechanics analysis assuming that the vessel material may behave in a non-ductile manner. In the temperature region characterized by the Charpy lower shelf and transition region a LEFM analysis is required using the procedure described in Appendix G of 10 CFR 50. In the Charpy upper shelf temperature region no additional fracture mechanics analysis is required as long as it is demonstrated that the Charpy upper shelf energy is greater than 50 ft-lbs. If the Charpy energy is predicted to drop below the 50 ft-lb level, it is required to provide supplemental fracture toughness information and an analysis to demonstrate an equivalent margin of safety as required by Appendix G of 10 CFR 50. This necessitates the use of elastic-plastic fracture mechanics analysis methods.

As part of the required supplemental analysis, a criteria must be established such that a smooth transition will occur in the vessel operating limitations between the required LEFM analysis and the supplemental EPFM analysis. A conservative approach to establishing this transition is to perform both LEFM and EPFM analyses and establish the vessel operating limits as the lower bound of the two results.

The temperature at which the transition is made from the LEFM analysis to the EPFM analysis is therefore defined as the temperature at which the allowable pressure versus temperature curves calculated by the two procedures intersect. Since the allowable pressure versus temperature curve obtained from the EPFM analysis is based on a structural instability analysis which is a function of both the structure's geometry and the material properties, the temperature at which this transition is made in general is not a function of material properties alone (see section 6).

For the specific case where the J-R curve is obtained from small RVSP fracture toughness specimens, the temperature can be determined from the material properties data alone. Because of the limited crack extension and limitations on the maximum J values allowed by ASTM, the J value at calculated instability will always be the maximum J value measured on the surveillance specimen. This limitation imposed by the specimen size provides additional conservatism in the EPFM analysis since the applied J value to cause instability of the structure will always be greater than the maximum J value obtained from the surveillance specimens.

The temperature at which the transition is made from LEFM to EPFM can be obtained by the procedure shown schematically in Figure 3-1. The K_{JR} in this figure is obtained using the proced[illegible]or converting from J to K values found in ASTM E813.

This procedure for determining the temperature for the transition from LEFM to EPFM will always be conservative because it is based on the K_{IR} curve. Since the K_{IR} curve is based on dynamic fracture tests (both dynamic loading and crack arrest), it is impossible for cleavage fracture to occur at temperatures greater than those obtained using this procedure and the K_{IR} curve.

Table 3-1. Summary of RT_{NDT} Data and Estimated Temperatures

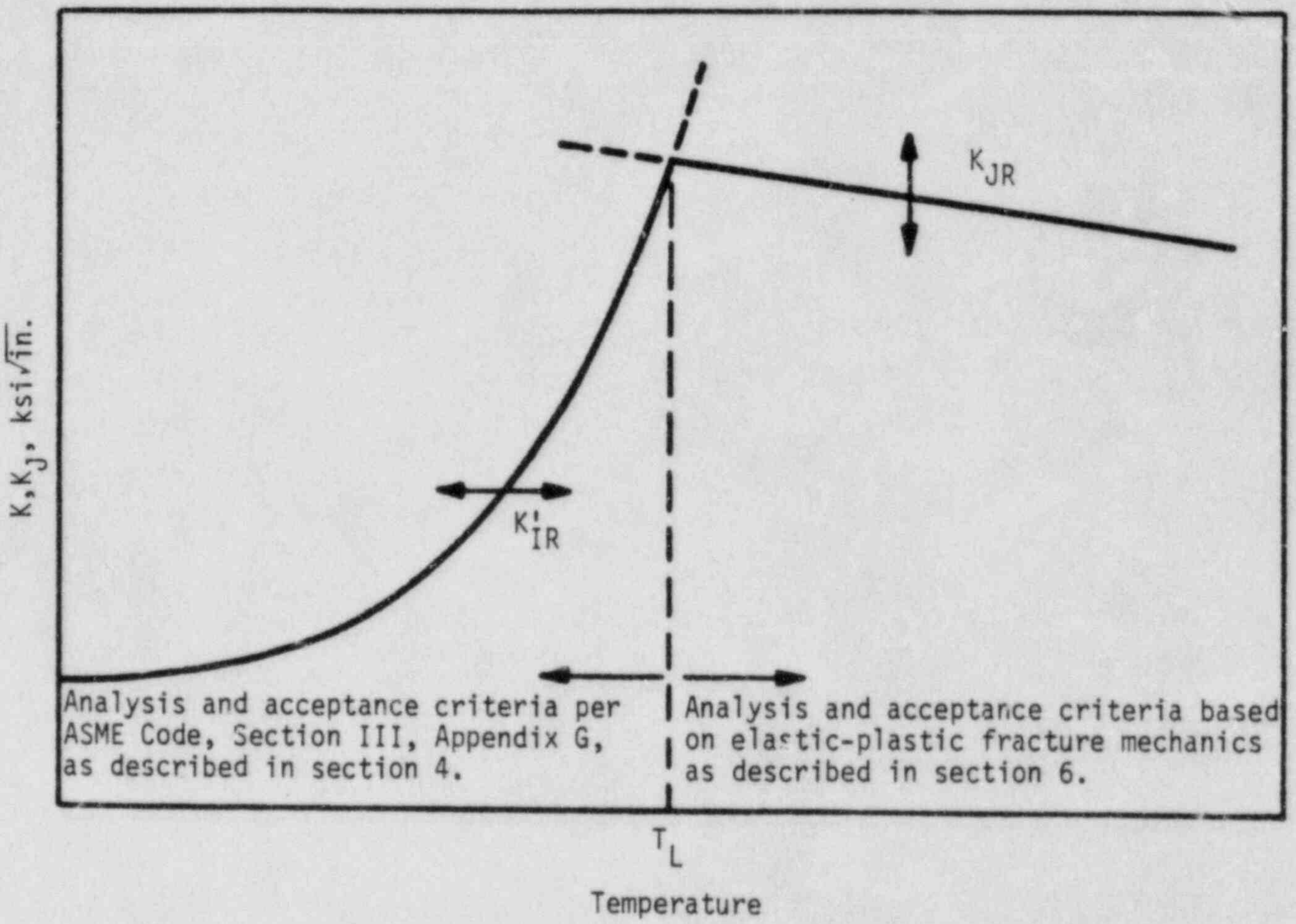
Material/type	No. of cases	RT_{NDT}, F High meas	Avg. meas	Est	Diff between ave. measured and estimated RT_{NDT},F
SA 508, Class 2 low-alloy forgings	24	60	4	60 or T_{NDT}[(a)]	56
SA 533 B low-alloy plates	13	40	0	40	40
SA 516 C carbon steel plates	20	10	-11	10	21
Submerged-arc Linde 80 weld	10	20	0	20	20
Submerged-arc Linde 0091 weld	10	-50	-66	20	86
Manual metal arc weld	9	-10	-67	20	87
SA 508 Class 2 HAZ	6	30	-25	30	55
SA 533 B HAZ	11	10	-23	10	33
SA 516 C HAZ	7	-20	-26	-20	6
SA 106 C piping	11	50	5	50	45

(a) 60F or the drop weight temperature, if known.

Table 3-2. Summary of C_vUSE Data and Estimated Upper Shelf Energies

Material/type	No. of cases	C_vUSE, ft-lb Low meas	Avg. meas	Est	Diff between ave. measured and estimated C_v USE, ft-lb
SA 508, Class 2 low-alloy forgings	5	91	124	75	49
SA 533B low-alloy plates	8	85	91	75	16
Submerged-arc weld	20	66	81	66	15

Figure 3-1. Relationship Between Fracture Toughness Properties and the Fracture Mechanics Evaluation Methods



Legend

K'_{IR} -- K_{IR} relationship as defined in ASME Code, Section III, Appendix G, for a specific material and adjusted for initial properties and effects of neutron irradiation.

K_{JR} -- Materials elastic-plastic fracture toughness relationship as developed from appropriate data base.

T_L -- Temperature at which the linear-elastic fracture mechanics and elastic-plastic fracture mechanics analytical methods interface.

4. LEFM ANALYTICAL PROCEDURES

4.1. Basis

The calculational procedures used to determine the pressure-temperature limitations on the reactor coolant (RC) system are based on ASME Appendix G, as incorporated in the Winter 1973 Addenda, and on WRC Bulletin 175.[3] To determine the minimum bolt preload temperature, the calculational procedure is partially based on Appendix A to ASME Section XI since it uses the static critical stress intensity factor K_{Ic} rather than the reference critical stress intensity K_{IR} of ASME Appendix G.

Procedures for quantitatively obtaining the maximum allowable pressure at a given temperature for Class 1 ferritic pressure-retaining components are given in ASME Appendix G and are described in more detail in WRC Bulletin 175. The methods of calculating applied stress intensity factor are simplified, and the postulated flaw is defined by a reference flaw of specified size and shape. The procedures are not applicable to pressure boundary regions near geometric discontinuities, such as nozzles, and in such cases the technology of Bulletin 175 is applied directly.

The components of the RC system in a typical B&W power plant have been analyzed to determine the minimum required reactor coolant temperature for pressures of 626, 2250, and 3125 psig. The 626 psig pressure was selected because it is 1 psig above the pressure corresponding to 20% of the preoperational system hydrostatic test pressure. This is the maximum allowable pressure (625 psig) for a component when the reactor coolant temperature (or the volumetric average metal temperature) is below the lowest service temperature of the component. The components for which a lowest service temperature must be defined include the RC loop piping and the control rod drive mechanism (the CRDM is an appurtenance to the reactor vessel). The lowest service temperature of these components is 150F (based on

RT_{NDT} + 100F) for the piping and 100F (as derived in section 3.3) for the CRDM. The 2250 psig pressure was selected because it is approximately the normal operating pressure; 3125 psig was selected because it is the preservice system hydrostatic test pressure.

The reactor vessel closure head region, the reactor vessel outlet nozzles, and the beltline region are the only portions of the RC system with a relatively high minimum required temperature at 626 and 2250 psig. The reactor vessel outlet nozzle and the closure head region show the highest minimum required temperature at 3125 psig. These three regions are the only ones that, at different stages of the vessel's design life, regulate the pressure-temperature limitations of the RC system for normal operation and inservice pressure tests. The outlet nozzles and the closure head region regulate the minimum allowable preservice hydrostatic test temperature. Each region has the following characteristics:

The beltline region directly surrounds the effective height of the fuel assemblies and is exposed to continual neutron flux throughout the service life of the reactor vessel. The neutron fluence (flux x time) will change the mechanical properties of the beltline region materials. This continual change necessitates periodic adjustments to the pressure-temperature operating limitation throughout the service life of the reactor vessel. This region is remote from geometric discontinuities, and the applied stresses are proportional to the internal pressure and to the heatup or cooldown rates.

The closure head region of the reactor vessel is subject to significant stresses due to mechanical loads resulting from bolt preload. In this region, the applied stresses are not proportional to the internal pressure. This region is subjected to high stresses at relatively low temperatures. The highest stress levels occur at the head-to-head flange juncture of the closure head region.

The outlet nozzle of the reactor vessel is the largest nozzle in the RC system. The inside corner of the nozzle is subjected to high local stresses produced by pressure. The local stresses can be two to three times the membrane stress of the shell. As the radius of the nozzle increases, the magnitude of the stress intensity factor increases for a constant assumed flaw.

For loading conditions other than the preservice system hydrostatic test (PSHT), the nozzles and most other regions near geometric discontinuities are analyzed using the same safety margins as those required by ASME Appendix G for shells and heads remote from discontinuities. Margins of safety are imposed on the stress intensity safety factors and the postulated flaw size. For the analysis of the head-to-head flange juncture of the closure head region, the stress intensity safety factors are the same; however, the size of the postulated flaw is smaller than the referenced flaw. The assumed flaw on the head-to-head flange juncture is a sharp surface flaw with a depth of 1/6 t and a length of t (where t is the section thickness). The thickness of the juncture varies from 6.5 to 8 inches depending on the size of the reactor vessel. This juncture is inspected prior to service and at several intervals throughout the service life of the power plant. The inspection techniques can detect very small surface defects (defects with areas greater than 1 in.^{2} are considered detectable). For the wall thickness of 6.5 inches, the area of the postulated flaw (semi-elliptical) is 8.5 in.^{2}. The area of the postulated flaw is 8.5 times larger than the minimum detectable defect area.

For the PSHT all geometries are analyzed using a margin of safety of 1.0 on the stress intensity factor and postulated flaws that are smaller than the reference flaw of ASME Appendix G. Smaller postulated flaws are justifiable since this test is performed before initial operation. The postulated flaws employed to determine the pressure-temperature limit curve for the PSHT are described in section 4.2.2.2. Additionally a pressure exceeding 2/3 of the test pressure is not allowed until the component temperature exceeds RT_{NDT} + 60.

The reference flaw of ASME Appendix G is a sharp surface flaw perpendicular to the direction of maximum stress, having a depth of 1/4 t and length of 1-1/2 t (for section thicknesses of 4 to 12 inches). ASME Section III also requires that the test coupons be at least 1/4 t from any surface unless the material is a very thick forging and the test location is very near the surface (0.75 inch from a heat-treated surface). Since for most geometries the depth of the postulated flaw and the test location is 1/4 t (from either surface), the analytical calculations used on all geometries depend on the metal temperature and impact properties (including effects of irradiation) at 1/4 t and 3/4 t. The impact properties of thick and complex forgings at 1/4 and 3/4 t are assumed to be equal to the properties determined near the surface. The metal temperature and impact properties for the head-to-head flange juncture are taken at 5/6 t. For the analysis of the head-to-head flange juncture, the impact properties at 5/6 t are assumed to be equal to those determined by the ASME Code.

At the beginning of service life, the closure head region and the outlet nozzles control the pressure-temperature limitations of the loading conditions of interest. After several years of neutron irradition exposure, the RT_{NDT} of the beltline region materials will be high enough for the beltline region to regulate parts of the pressure-temperature limit curves. The maximum allowable pressure as a function of fluid temperature for the service period of the limit curves is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzle, and beltline region. The maximum allowable pressure is the lower of the three calculated pressures. For additional years' operation, the adjusted RT_{NDT} of the beltline region materials will continue to increase; therefore, periodic adjustments on the pressurization limit curves are required throughout the service life of the RC system. Since every surveillance capsule withdrawal will produce pertinent irradiated beltline region material impact data, adjustment of the pressurization limit curves may be required after each capsule withdrawal. The initial and subsequent adjusted pressure-temperature limits include the predicted radiation-induced RT_{NDT} (determined as described in section 3.1.3.1) for the period until the next capsule withdrawal.

After each capsule withdrawal, the RT_{NDT}s of the beltline region materials are predicted by adding the unirradiated values to the predicted radiation-induced ΔRT_{NDT}s and then confirmed by the material surveillance program test results. Both the predicted ΔRT_{NDT} and the data obtained from the surveillance program are used to define the adjusted RT_{NDT} that will be used to recalculate the pressurization limit curves.

4.2. Description

The methods used to obtain the pressure-temperature limitations for each of the loading conditions of interest are described in this section. Table 4-1 summarizes the analytical assumptions.

4.2.1. Normal Operation

4.2.1.1. Bolt Preloading

To define the minimum preload temperature, it is necessary to analyze the bolt preloading conditions. The minimum preload temperature can be the lowest temperature at which the bolting materials meet the toughness requirements of the ASME Code or the calculated minimum temperature required for protection against nonductile failure of the closure head region, whichever temperature is higher. Section 3.2 of this report shows that at 40F the bolting materials meet the requirements of ASME Code; now it is necessary to calculate the minimum allowable temperature of the closure head region to determine whether it is higher than 40F.

During bolt preloading, the maximum tensile stresses occur at the outside surface (5/6 t) of the head-to-head flange juncture of the closure head region. The stresses are primarily bolt preload bending stresses. The pressure stresses are very small since the maximum allowable pressure for this loading condition is relatively low (~450 psig). The minimum temperature required for protection against nonductile failure is first calculated at 0 psig and then at 626 psig. Both pressures are analyzed because higher temperatures may be required at 0 than at 626 psig. For both cases, the thermal stresses are nil since the coolant temperature is essentially at steady state throughout this loading condition. The method used to calculate the minimum preload temperature is as follows:

1. The membrane and bending stresses at the 5/6 t vessel wall location that result from bolt preload and internal pressure are calculated by the stress analysis of the head-to-head flange juncture at both 0 and 625 psig.
2. Using the membrane and bending stresses calculated in step 1, the stress intensity factor for both cases is calculated by the following equation:

$$K_I = 1.1\ \sigma_m M_K \frac{\sqrt{\pi a}}{\sqrt{Q}} + \sigma_b M_B \frac{\sqrt{\pi a}}{\sqrt{Q}}$$

where the assumed flaw of a = 1/6 t;

$$K_I = 0.82\ \sigma_m \frac{\sqrt{t}}{\sqrt{Q}} + 0.64\ \sigma_b \frac{\sqrt{t}}{\sqrt{Q}}$$

where K_I = stress intensity factor based on reference flaw at 5/6 t vessel wall location,

M_K, M_B = correction factors for membrane and bending load conditions, respectively (values from WRC Bulletin 175, Figures A3-1 and A3-2); for a 1:6 crack depth: thickness ratio the values are 1.03 and 0.88, respectively;

Q = flaw shape factor modified for plastic zone size (reference 3 gives basic expression)

a = assumed crack depth,

t = section thickness,

σ_m = calculated membrane stress,

σ_b = calculated bending stress.

3. The relative temperature T-RT_{NDT} at which the critical static stress intensity factor K_{Ic} equals the highest calculated stress intensity factor K_I (from step 2) is calculated using Figure 4-1, which is based on Figure A-4200-1 from ASME XI, Appendix A.
4. Using the relative temperature calculated in step 3 and the highest RT_{NDT} of the closure head region materials, we can calculate the minimum temperature required for protection against nonductile failure.
5. The minimum preload temperature is the one calculated in step 4 or 40F, whichever is higher.

4.2.1.2. Heatup

The heatup transient starts at the minimum preload temperature. For temperatures above minimum preload, the heatup pressure-temperature limit curve is calculated by a point-by-point comparison of the limits imposed by the closure head region, the outlet nozzles, and the beltline region. The heatup limit curve is the composite or lower bound curve of the limits imposed by the three controlling regions.

The limits imposed by the closure head region are established by assuming a 1/6 t x t surface flaw located at the outside surface of the head-to-head flange juncture. During heatup all the stresses, including the bolt preload and the thermal stresses, are in tension at the outside surface of the closure head region. The 5/6 t location corresponds to the depth of the assumed flaw on the outside surface of the head-to-head flange juncture. The minimum required fluid temperatures are calculated at several coolant pressures above 625 psig. This is done by first calculating the fluid temperature as a function of metal temperatures for each heatup rate of interest and then calculating the minimum required metal temperature at each pressure. For fluid temperatures between the minimum preload temperature and the minimum required fluid temperature at 626 psig, the maximum allowable pressure is 625 psig.

The limits imposed by the outlet nozzles are calculated by assuming a flaw at the inside corner of the nozzle. The depth of the assumed flaw is 3 inches, which is the depth of the reference flaw of ASME Appendix G for a section thicker than 12 inches. During heatup, the inside corners of nozzles are subjected to high local stresses produced by pressure; however, the thermal stresses are in compression. The limit curve is calculated by determining the metal temperature at the inside corner 1/4 t of the outlet nozzle as a function of fluid temperature. The critical stress intensity factor is indexed to the fluid temperature using the highest RT_{NDT} of the two outlet nozzles. The maximum allowable pressure is then calculated as a function of fluid temperature. The thermal stress intensity factors for these calculations are assumed to be zero. This assumption is conservative since during heatup, the contributing thermal stress intensity factor at the inside corner of the nozzle is negative.

The pressure-temperature limits imposed by the beltline region are calculated using the postulated reference flaw of 1/4 t depth. The reference flaw is assumed to be located at both the inside and outside surfaces of the beltline region. During heatup, the thermal stresses are in compression at 1/4 t of the section thickness of the beltline region and are in tension at 3/4 t. The 1/4 t location corresponds to the depth of the reference flaw on the inside surface of the reactor vessel wall. The 3/4 t location corresponds to the depth of the reference flaw on the outside surface of the reactor vessel wall. The metal temperatures at the 1/4 and 3/4 t lag the fluid temperature during the normal heatup conditions. Since the neutron fluence attenuates through the thickness of the beltline region material, the RT_{NDT} at the 1/4 t location will be higher than that at the 3/4 t location. Because of these variables, two sets of calculations must be performed to obtain the pressure-temperature limitations imposed by the beltline region.

First, the pressurization limit for the steady-state condition is calculated as a function of fluid temperature. For this calculation the metal and fluid temperatures are the same and the impact properties used are those of the 1/4 t location. There are no thermal stresses in this case, and the only contributing stress intensity factors are those produced by pressure.

Second, the curve of pressure versus fluid temperature limit is calculated for each heatup ramp of interest assuming that the reference flaw is located at the outside surface of the beltline region wall. For this calculation, it is necessary to determine the metal temperature at 3/4 t as a function of fluid temperature and the stress intensity factor produced by the thermal stresses. The thermal stress intensity factor is added to the pressure stress intensity factor. The impact properties used in this calculation are those of the 3/4 t vessel wall location.

The methods employed to obtain the limits imposed by the closure head region, outlet nozzle, and beltline region and the pressure-temperature limit curve of the RCPB for normal heatup are described below.

<u>Closure Head Region Heatup Limits</u>

The heatup limits imposed by the closure head region are calculated as follows:

1. For each of the heatup ramps of interest, the metal temperature at 5/6 t of the head-to-head flange juncture is calculated as a function of fluid temperature.

2. The minimum allowable fluid temperatures of the closure head region for coolant pressures of 626, 1250, and 2250 psig are calculated as follows:

a. The membrane and bending stresses at the 5/6 t location resulting from bolt preload, thermal gradient, and internal pressure are calculated by a detailed stress analysis of the head-to-head flange juncture.

b. Using the membrane and bending stresses calculated in step a, the stress intensity factor is calculated by the following equation:

$$K_I = 2\left[1.1(\sigma_{mb} + \sigma_{mP})M_K \frac{\sqrt{\pi a}}{\sqrt{Q}} + \sigma_{bb}M_B \frac{\sqrt{\pi a}}{\sqrt{Q}}\right] + \sigma_{bT}M_B \frac{\sqrt{\pi a}}{\sqrt{Q}}$$

where the assumed flaw of a = 1/6 t;

$$K_I = 1.64(\sigma_{mb} + \sigma_{mP}) \frac{\sqrt{t}}{\sqrt{Q}} + 1.28\,\sigma_{bb} \frac{\sqrt{t}}{\sqrt{Q}} + 0.64\sigma_{bT} \frac{\sqrt{t}}{\sqrt{Q}}$$

where σ_{mb}, σ_{mP} = calculated membrane stresses due to bolt preload and pressure,

σ_{bb}, σ_{bT} = calculated bending stresses due to bolt preload and thermal gradient. (See section 4.2.1.1 for definition of other factors.)

c. For each pressure, the minimum relative temperature is that at which the calculated stress intensity factor (K_I) equals the reference stress intensity factor (K_{IR}) of Figure G-2110.1 of ASME Appedix G.

d. The minimum required metal temperatures are calculated using the minimum relative temperatures (calculated in step c) and the highest RT_{NDT} of the closure head region materials.

e. The minimum allowable fluid temperature for the three coolant pressures are calculated using the minimum required metal temperatures calculated in step d and the fluid-metal temperature relationship of step 1.

3. The pressure-temperature limits imposed by the closure head region during normal heatup are defined as follows:

a. For fluid temperatures between the minimum preload temperature calculated in section 4.2.1.1 and the minimum allowable fluid temperature calculated in step 2 for 626 psig, the maximum allowable coolant pressure is 625 psig.

b. For pressures of 1250 and 2250 psig, the minimum allowable fluid temperatures are those calculated in step 2. For coolant pressures between 626, 1250, and 2250 psig, the minimum fluid temperatures are defined by linear interpolation.

Outlet Nozzle Heatup Limits

The heatup limits imposed by the outlet nozzles are calculated as follows:

1. For each of the heatup ramps of interest, the metal temperature at a depth of 3 inches (at the inside corner) location of the outlet nozzle is calculated as a function of fluid temperature. The thermal analysis calculations are performed using a one-dimensional transient distribution program.

2. The K_{IR} curve of ASME Appendix G is indexed to the highest RT_{NDT} of the two outlet nozzles. Using the fluid-metal temperature relationship calculated in step 1, the critical stress intensity factor is calculated as a function of fluid temperature, $K_{IR}(T_f)_{3"}$.

3. The pressure-temperature limit curve imposed by the outlet nozzles during heatup is calculated using the following equation:

$$P(T_f) = \frac{K_{IR}(T_f)_{3"}}{2F(a/r_n)\,\dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}\,\sqrt{\pi a}}$$

where $K_{IR}(T_f)_{3"}$ = critical stress intensity factor as function of fluid temperature calculated in step 2,

$F(a/r_n)$ = obtained from WRC Bulletin 175, Figure A5-1,

r_n = apparent nozzle radius,

$P(T_f)$ = coolant pressure as function of fluid temperature,

r_o, r_i = outside and inside radius of reactor vessel nozzle belt,

a = flaw depth, assumed to be 3 inches.

Beltline Region Heatup Limits

The limits imposed by the beltline region are calculated as follows:

1. For each heatup ramp of interest, the metal temperatures at the 3/4 t vessel (beltline region) wall location are calculated as a function of fluid temperature (T_f). The thermal analysis calculations are performed by a one-dimensional transient distribution program.

2. Also, as part of the thermal analysis of step 1 in the preceding paragraph, the temperature distribution through the vessel (beltline region) wall is calculated as a function of fluid temperature for each heatup ramp.

3. The K_{IR} curve of ASME Appendix G (Figure G-2110.1) is indexed to the highest postulated RT_{NDT} of the 1/4 t wall location and to the highest postulated RT_{NDT} of 3/4 t. For each heatup ramp of interest the critical stress intensity factor for the 3/4 t vessel wall location is calculated as a function of fluid temperature using the data of step 1 and the RT_{NDT} at 3/4 t. Also, for the steady-state condition, K_{IR} is plotted as a function of fluid temperature using the RT_{NDT} at 1/4 t.

4. The K_I produced by the thermal gradient across the vessel wall is calculated as follows:

   a. Utilizing the temperature distribution obtained in step 2 the equivalent linear bending stress is calculated due to the radial gradient. This is done by either integrating the thermal distribution or stress distribution across the wall.

   b. The $K_{IT} = M_b \times Sth$ where M_b equals 2/3 Mm as defined in ASME Appendix G and Sth is the equivalent linear thermal bending stress.

5. The pressurization limit for a steady-state condition is calculated as a function of fluid temperature by the following equation:

$$P(T_f)_{ss} = \frac{K_{IR}(T_f)1/4\ t}{2M_m \dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}}$$

where $K_{IR}(T_f)1/4\ t$ = critical stress intensity factor for steadystate condition as a function of fluid temperature, based on RT_{NDT} at 1/4 t, calculated in step 4;

M_m = obtained from ASME Appendix G, Figure G-2214.1; a stress ratio $\geq$ actual is used (Checks are made to confirm that the proper M_m value is used.);

r_i, r_o = inside and outside radii of reactor vessel beltline region,

$P(T_f)_{ss}$ = allowed steady-state pressure as a function of fluid temperature.

6. The pressure versus fluid temperature data for each heatup ramp of interest are calculated as follows:

$$P(T_f) = \frac{K_{IR}(T_f)3/4\ t - K_{IT}(T_f)}{2M_m \dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}}$$

where $K_{IR}(T_f)3/4\ t$ = critical stress intensity factor based on 3/4 t RT_{NDT}, a function of fluid temperature calculated in step 4,

$K_{IT}(T_f)$ = K_I produced by thermal gradient across the vessel wall as a function of fluid temperature (calculated in step 5)

$P(T_f)$ = allowable pressure as a function of fluid temperature,

and the other factors are as defined above.

7. The pressure-temperature limits imposed by the beltline region during normal heatup are obtained by a point-by-point comparison of the data obtained in steps 5 and 6. The maximum allowable pressure is taken to be the lower of the two values.

Reactor Coolant Pressure Boundary Heatup Limits

The pressure-temperature limits during normal heatup of the RCPB are obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure at any given fluid temperature is taken to be the lower of the three calculated pressures.

4.2.1.3. Cooldown

The method used to obtain the cooldown pressurization limit curve for the RCPB is very similar to that used for the heatup curve. From the normal operating temperature to the minimum bolt preload temperature, the cooldown pressure-temperature limit curve is calculated through a point-by-point comparison of the limits imposed by the closure head region, the outlet nozzles, and the beltline region. The cooldown limit curve is the lower bound curve of the limits imposed by the three controlling regions.

The cooldown limits of the closure head region are established, as for heatup, by assuming a 1/6 t x t surface flaw located at the outside surface of the head-to-head flange juncture. Although the inside surface is subjected to positive thermal stresses during cooldown, the total stress is higher at the outside than at the inside surface. This is due to the high bolt preload bending stresses on the outside surface. The cooldown and heatup limits of the closure head region are calculated very similarly. The only differences are that (1) the fluid and metal temperatures are assumed to be equal (steady-state), and (2) the thermal stresses at the outside surfaces are assumed to be zero. The steady-state assumption is conservative since the metal temperature, especially at 5/6 t, is higher than the fluid temperature during cooldown. The assumption that the thermal stresses are zero is also conservative since the thermal stresses at the outside wall of the closure head region are negative during cooldown.

The cooldown limits imposed by the outlet nozzles are calculated, as for heatup, assuming a 3-inch-deep flaw at the inside corner of the nozzle. During cooldown, the inside corners of the nozzles are subjected to high local stresses produced by the pressure and temperature gradient. To calculate the limit curve, the metal temperature 3 inches from the inside corner locations is calculated as a function of fluid temperature. When calculating the maximum allowable pressure, the contributing thermal stress intensity factor is assumed to be equal to that calculated for the nozzle belt vessel wall. This assumption is conservative because the thermal stress intensity factor for a nozzle corner flaw is also lower than that for a surface flaw on the nozzle vessel wall owing to the lower postulated crack penetration (crack depth over section thickness) on the nozzle corner.

The method used to calculate the cooldown pressure limit curve imposed by the beltline region is also similar to that used for the heatup limit curve; however some differences exist. During cooldown, the thermal stresses are in tension at 1/4 t and in compression at 3/4 t. Because the thermal stresses are in tension at 1/4 t, and the RT_{NDT} at 1/4 t will be higher than that at 3/4 t after exposure to neutron irradiation, only the metal temperature and the impact properties of the 1/4 t location are used to obtain the cooldown limit curve. However, three calculational steps are required to obtain the cooldown limit curve of the beltline region:

1. The pressure limit curve for a steady-state condition is calculated as a function of fluid temperature. The assumed steady-state condition makes the fluid and metal temperatures equal. The impact properties are those of the 1/4 t location. The contributing thermal stress intensity factor is zero. This step is required because the metal temperature may not be higher than that of the fluid during an upset cooldown condition as it is during normal cooldown.
2. The pressure limit curve is calculated for each cooldown ramp of interest assuming that the reference flaw is located at the inside surface of the beltline region wall. For this calculation, the metal temperature at 1/4 t is determined as a function of fluid temperature, and the thermal stress intensity factor is added to the stress intensity factor produced by pressure. The impact properties at 1/4 t are used in this calculation.
3. A point-by-point comparison of the data obtained in the first two steps will obtain the lowest pressure at any temperature of the two data sets. The calculated lowest pressure becomes the maximum pressure at any temperature for the reactor vessel beltline region.

The methods used to obtain the limits imposed by the closure head region, outlet nozzle, and beltline region and the pressure-temperature limit curve for the RCPB for normal cooldown are described below.

Closure Head Region Cooldown Limits

The cooldown limits imposed by the closure head region are calculated as follows:

1. For each cooldown ramp of interest, the metal temperature at 5/6 t of the head-to-head flange juncture is assumed to be equal to the fluid temperature.

2. The minimum allowable fluid temperatures of the closure head region for pressures of 626, 1250, and 2250 psig are calculated as follows:

   a. The membrane and bending stresses at 5/6 t resulting from bolt preload and internal pressure are calculated by a detailed stress analysis of the head-to-head flange juncture.

   b. Using the membrane and bending stresses calculated in step a, the stress intensity factor is calculated using the following equation:

   $$K_I = 2\left[1.1(\sigma_{mb} + \sigma_{mp})M_K\frac{\sqrt{\pi a}}{\sqrt{Q}} + \left(\sigma_{bb}M_B\frac{\sqrt{\pi a}}{\sqrt{Q}}\right)\right]$$

   where the assumed flaw of a = 1/6 t;

   $$K_I = 1.64(\sigma_{mb} + \sigma_{mp})\frac{\sqrt{t}}{\sqrt{Q}} + 1.28\ \sigma_{bb}\frac{\sqrt{t}}{\sqrt{Q}}$$

   where K_I, M_K, M_B, Q, a, and t are defined in section 4.2.1.1 and other factors in "Closure Head Region Heatup Limits," step 2b.

   c. For each pressure the minimum relative temperature is that at which the calculated stress intensity factor K_I equals the reference stress intensity factor (K_{IR} of ASME Appendix G, Figure G-2110.1).

   d. The minimum required fluid temperatures are calculated using the minimum relative temperatures (calculated in step c) and the highest RT_{NDT} of the closure head region materials.

3. The pressure-temperature limits imposed by the closure head region during normal cooldown are defined as follows:

a. For fluid temperatures between the minimum preload temperature calculated in section 4.2.1.1 and the minimum allowable fluid temperature calculated in step 2 for 626 psig, the maximum allowable pressure is 625 psig.

b. The minimum allowable fluid temperatures for pressures of 1250 and 2250 are those calculated in step 2. For coolant pressures between 625, 1250, and 2250 psig, the minimum fluid temperatures are defined by linear interpolation.

Outlet Nozzle Cooldown Limits

The cooldown limits imposed by the outlet nozzles are calculated as follows:

1. For each cooldown ramp of interest, the metal temperature at the 3-inch depth (from the inside corner) location of the outlet nozzle is calculated as a function of fluid temperature. The thermal analysis is performed using a one-dimensional transient distribution program.
2. As part of the thermal analysis in step 1, the temperature difference through the nozzle belt vessel wall is calculated as a function of fluid temperature.
3. The K_{IR} curve of ASME Appendix G is indexed to the highest RT_{NDT} of the two outlet nozzles. Using the data calculated in step 1, the critical stress intensity factor at the inside corner of the nozzle is calculated as a function of fluid temperature, $K_{IR}(T_f)_{3"}$.
4. The K_I produced by the thermal gradient across the outlet nozzle corner is calculated by same method as step 4 in heatup procedure.
5. The pressure-temperature limit curve imposed by the outlet nozzles during cooldown is calculated using the following equation:

$$P(T_f) = \frac{K_{IR}(T_f)_{3"} - K_{IT}(T_f)}{2F(a/r_n)\,\dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}\,\sqrt{\pi a}}$$

where $K_{IR}(T_f)_{3}$" = critical stress intensity factor as a function of fluid temperature calculated in step 3,

$K_{IT}(T_f)$ = thermal stress intensity factor as a function of fluid temperature calculated in step 4,

and all other factors are defined in "Outlet Nozzle Heatup Limits," step 3.

Beltline Region Cooldown Limits

The limits imposed by the beltline region during cooldown are calculated as follows:

1. For each cooldown ramp of interest, the temperature at 1/4 t (beltline region) is calculated as a function of fluid temperature (T_f).
2. As part of the thermal analysis of step 1, the temperature difference through the vessel (beltline region) wall is also calculated as a function of fluid temperature for each cooldown ramp.
3. The most limiting adjusted RT_{NDT} at 1/4 t is also used in the cooldown analysis.
4. The K_{IR} curve of ASME Appendix G is indexed to the adjusted RT_{NDT} of step 3. For each cooldown ramp of interest, K_{IR} is plotted as a function of fluid temperature using the data from step 1. For the steady-state condition, K_{IR} is also plotted as a function of fluid temperature using the same adjusted RT_{NDT}.
5. The K_I produced by the thermal gradient across the vessel wall during cooldown is calculated as described in step 4 of the heatup procedure. However, the ΔT values are those calculated in step 2 for each cooldown ramp.
6. The pressurization limit for steady-state condition is calculated as a function of fluid temperature, as described in "Beltline Region Heatup Limits," step 5.
7. The pressure-versus-fluid temperature data for each cooldown ramp are calculated as follows:

$$P(T_f) = \frac{K_{IR}(T_f)1/4\ t - K_{IT}(T_f)}{2M_m \dfrac{r_i^2 + r_o^2}{r_o^2 - r_i^2}}$$

where $K_{IR}(T_f)1/4\ t$ = K_I based on RT_{NDT}, also a function of fluid temperature (see step 4),

$K_{IT}(T_f)$ = thermal stress intensity factor as a function of fluid temperature (see step 5).

and the other factors are as defined in "Beltline Region Heatup Limits," steps 6 and 7.

8. The pressure-temperature limits imposed by the beltline region during normal cooldown are obtained through a point-by-point comparison of the data obtained in steps 6 and 7; the maximum allowable pressure is taken to be the lower of the two values.

Reactor Coolant Pressure Boundary Cooldown Limits

The pressure-temperature limits during normal cooldown of the RCPB are obtained through a point-by-point comparison of the limits imposed by the closure head region ("Closure Head Region Cooldown Limits," step 3), the outlet nozzles ("Outlet Nozzle Cooldown Limits," step 5), and the beltline region ("Beltline Region Cooldown Limits," step 8). The maximum allowable presure at any given fluid temperature is taken to be the lowest of the three calculated pressures.

4.2.2. Preservice System Hydrostatic Test (PSHT)

4.2.2.1. Bolt Preloading

The minimum preload temperature for the PSHT is calculated by following the basic methods employed for normal operation (section 4.2.1.1). For the PSHT the minimum preload temperature is calculated using a postulated surface flaw 1/8 t deep and 3/4 t long (1/8 t x 3/4 t) located in the outside surface of the head-to-head flange juncture. This assumed flaw is smaller than that assumed during normal operation (1/6 t x t). The smaller flaw is conservative since the PSHT is performed after the nondestructive testing required by ASME Section III, and the system has not been subjected to cyclic loading.

For the smaller postulated flaw, the equation used to calculate the stress intensity factor (step 2) takes the following form:

$$K_I = 0.70\ \sigma_m \frac{\sqrt{t}}{\sqrt{Q}} + 0.57\ \sigma_b \frac{\sqrt{t}}{\sqrt{Q}}$$

where K_I is the stress intensity factor based on a 1/8 t x 3/4 t flaw, and all other factors are as defined in section 4.2.1.1.

The values of σ_m and σ_b are calculated as described for normal operation (step 1) for the higher specified preload. All other steps of the procedure for calculating minimum preload temperature for normal operation are followed when calculating the minimum preload temperature for PSHT.

4.2.2.2. Heatup and Cooldown

As described in section 2.4, the PSHT pressure is normally reached when the metal temperature of the controlling pressure boundary is at steady state, and it is higher than the calculated minimum test temperature. At temperatures lower than this minimum, the maximum allowable pressure is only 625 psig. However, for some plants, it may be necessary to gradually increase the maximum allowable pressure as the metal temperature increases, just as for normal heatup and cooldown. For these plants the thermally induced stresses are considered when calculating the pressure-temperature limit curve. The methods for calculating the PSHT limit curve are similar to those for normal operation except for the following deviations:

1. The analysis is only performed for the two regions of the RCS that potentially control the PSHT pressure-temperature limits: the closure head region and the outlet nozzle. The beltline region does not control these limits since the materials have not been affected by irradiation.

2. When calculating the limits imposed by the closure head region, the postulated flaw is a 1/8 t x 3/4 t semi-elliptical surface flaw in the outside surface. The applied factor of safety in the stress intensity factor is 1.0, and the minimum allowable temperature is also calculated for 3125 psig. The postulated flaw is the same as that assumed when calculating the minimum preload temperature for the PSHT (section 4.2.2.1).

3. When calculating the limits imposed by the outlet nozzles, the postulated flaw is a surface flaw 1.0 inch deep located at the inside corner, and the factor of safety applied on the stress intensity factor due to pressure is 1.0. The justification for the smaller postulated flaw (1.0 rather than 3.0 inches deep) is again the nondestructive examination prior to PSHT and the impossibility of fatigue crack growth.

4. The pressure-temperature limits are calculated for both heatup and cooldown; however, for simplicity, the most limiting curve is used to define these limits from initiation to completion of the PSHT. The PSHT limit curve for the RCPB is the composite or lower bound of the limits imposed by the two controlling regions during both heatup and cooldown.

4.2.3. Inservice System Leak and Hydrostatic Tests (ISLHT)

4.2.3.1. Bolt Preloading

The minimum preload temperature for the ISLHT is the same as that for normal operation since the same load is specified.

4.2.3.2. Heatup and Cooldown

Since the ISLHT can be performed throughout the service life of the power plant, the effects of irradiation are considered when establishing the pressure-temperature limit curve for each test. As for normal heatup and cooldown, the closure head region, the outlet nozzles, and the beltline region are the only regions of the reactor vessel that control the pressurization limits of the RC system during ISLHT. The normal means of heating or cooling the system, before or after reaching the desired pressure for each test, are those used during normal heatup and cooldown. Consequently, the methods used to obtain the pressure limit curves of these loading conditions are similar to those used for normal heatup and cooldown. As for the PSHT, the ISHLT pressure-temperature limits are calculated for both heatup and cooldown; however, for simplicity, the most limiting curve is used to define the pressure-temperature limits from initiation to completion of the ISHLT. Another deviation from the methods employed for normal heatup and cooldown is the magnitude of the applied factor of safety. The factor of safety applied to calculate the stress intensity factor and the allowable pressure in the preceding procedures is 1.5 rather than 2.0. The ISLHT pressure-temperature limit curve is the composite or lower bound curve of the limits calculated for heatup and cooldown.

4.2.4. Reactor Core Operation

Except for low-power physics tests, the pressure-temperature limits for reactor core operation are as follows:

1. The fluid temperature must be equal to or higher than the minimum required for the ISLHT as calculated by the method described in section 4.2.3.
2. In addition, the fluid temperature must be at least 40F higher than the minimum pressure-temperature limit curve for both normal heatup and cooldown as calculated by the methods described in section 4.2.1.
3. The fluid temperature must be at least 525F.

These pressure-temperature limits for reactor core operation are in accordance with Appendix G to 10 CFR 50.

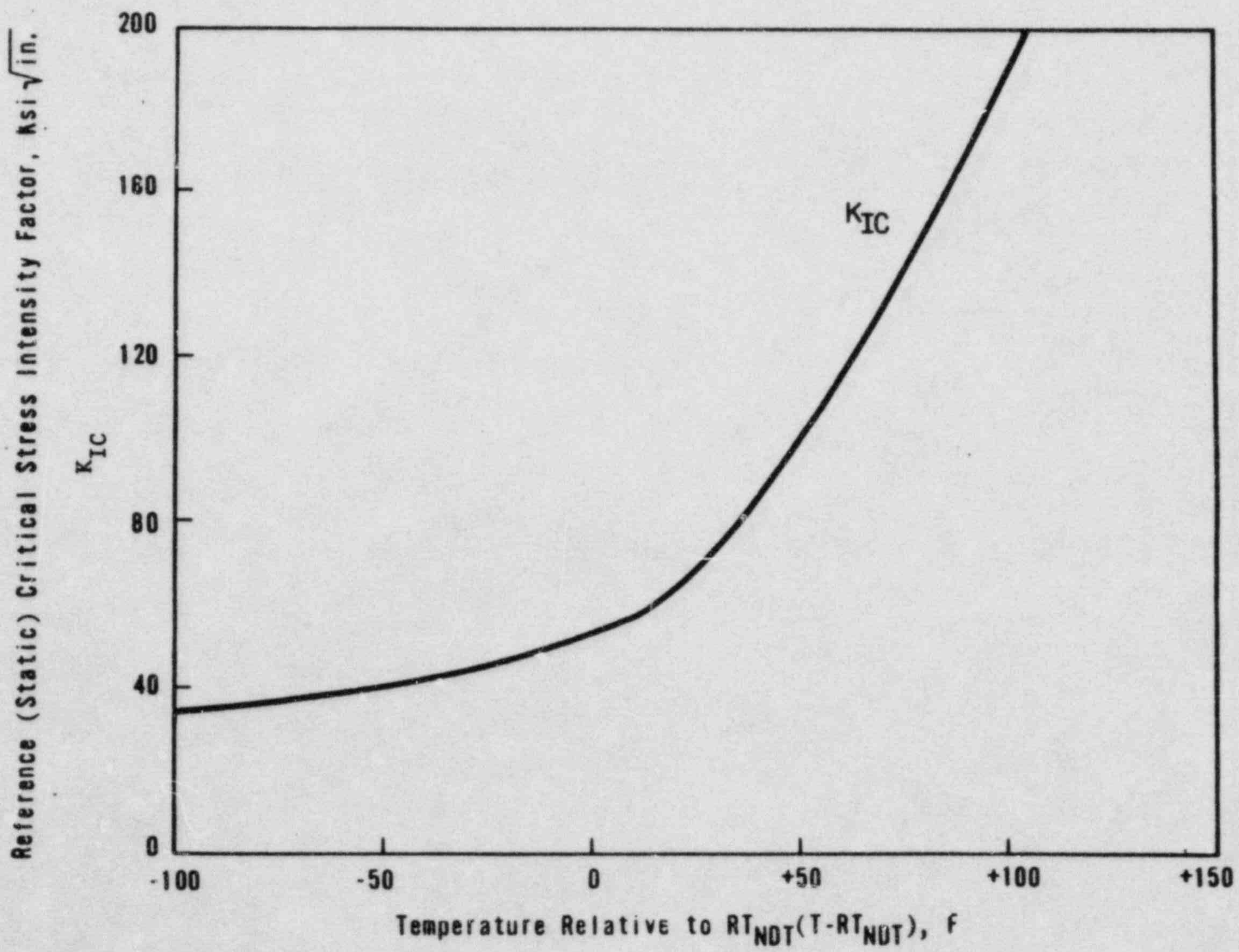
Table 4-1. Outline of Methods

Loading condition	Region analyzed	Flaw		Appl safety factor[a]			Temperature relationship	Mat'l property	
		Loc'n	Depth	K'_I	K''_I	K_{IT}		RT_{NDT}[b]	K_I
Normal bolt preload	Closure head	OD	1/6 t	1	1	--	Steady-state	1/6 t	K_{IC}
Normal heatup	Closure head	OD	1/6 t	2	1	1	$T_f(T_m)$	1/6 t	K_{IR}
	Outlet nozzle	ID	3 in.	2	--	--	$T_f(T_m)$	3/4 in.	K_{IR}
	Beltline	ID	1/4 t	2	--	--	Steady-state	1/4 t	K_{IR}
		OD	1/4 t	2	--	1	$T_f(T_m)$	3/4 t	K_{IR}
Normal cooldown	Closure head	OD	1/6 t	2	1	--	Steady-state	3/4 in.	K_{IR}
	Outlet nozzle	ID	3 in.	2	--	1	$T_f(T_m)$	3/4 in.	K_{IR}
	Beltline	ID	1/4 t	2	--	--	Steady-state	1/4 t	K_{IR}
		ID	1/4 t	2	--	1	$T_f(T_m)$	1/4 t	K_{IR}
Preservice SH test bolt preload, heatup and cooldown	Same as normal bolt preload, heatup and cooldown; however the depths of the postulated flaws are 1/8 t and 1.0 inch for the closure head region and outlet nozzle, respectively, the applied safety factor is always 1.0, and the beltline region is not considered. The limit curve is the composite of the limits imposed by the two controlling regions during both heatup and cooldown.								
Inservice SLH test bolt preload, heatup and cooldown	Same as normal bolt preload, heatup and cooldown, however, the applied safety factor is 1.5 rather than 2.0. The limit curve is the composite of the limits imposed by the three controlling regions during both heatup and cooldown.								

(a) K_I = stress intensity factor resulting from primary stresses, K" = stress intensity factor resulting from secondary stresses.

(b) Location of the RT_{NDT} used in the calculation.

Figure 4-1. Reference (Static) Critical Stress Intensity Factor Vs Temperature Relative to RT_{NDT} $(T-RT_{NDT})$



5. TYPICAL PRESSURE-TEMPERATURE LIMITS

5.1. Composite Limit Curves

The methods described in sections 3 and 4 have been applied to a typical 177 FA type plant to illustrate the development of the composite limit curves. The methods were applied for each of the loading conditions of interest. The analysis for normal heatup and cooldown was performed for the service periods ending at 5 and 32 effective full-power years (EFPY). The analysis for the inservice leak and hydrostatic tests (ISLHT) was performed for the service period ending at 5 EFPY. For consistency, the analysis was performed using a 100F/hour temperature ramp. For some transients the assumed 100F/hour ramp is not practical. The actual pressure-temperature limit curves for B&W plants may be different from those presented in this section because of the different maximum allowable temperature ramp rates and variations in ΔRT_{NDT}. The figures included here are for illustration only.

The analysis used the unirradiated impact properties, residual elements, predicted neutron fluence, and predicted radiation-induced ΔRT_{NDT} for the beltline region materials typical of an 177 FA-type plant. The unirradiated RT_{NDT}s of the closure head region materials and outlet nozzles are also those of a typical plant. The unirradiated impact properties and residual elements of the beltline region materials are listed in Table 5-1. The predicted neutron fluence values at the 1/4 t and 3/4 t beltline region locations for 5 and 32 EFPY and the corresponding ΔRT_{NDT}s and adjusted RT_{NDT} are listed in Table 5-2 for each of the beltline region materials.

Figures 5-1 and 5-2 illustrate the development of the composite pressure-temperature limit curves for a 100F/h normal heatup. The figures are applicable for the service periods ending at 5 and 32 EFPY, respectively. In addition to the composite limit curve, both figures show the limit curves imposed by the outlet nozzles, closure head region, and beltline region based on steady state and by the beltline region based on a finite heatup rate. As shown in Figure 5-1, the composite limit curve for 5 EFPY is the lower bound curve of the limits imposed by the outlet nozzles, closure head region, and beltline region based on a finite heatup rate. The composite limit curve for 32 EFPY is controlled by the limits set by the beltline region based on steady-state and finite heatup and the closure head region. At 5 EFPY, the limits set by the closure head region largely control the composite limit curve, and at 32 EFPY the same region only controls a small portion. This is because the limit curves set by the closure head region and outlet nozzles do not change throughout the service life of the power plant. Also, note that the limits set by the beltline region based on steady state do not control the composite limit curve for 5 EFPY, but they largely control the composite limit curve for 32 EFPY. This is due to the large difference in RT_{NDT} between 1/4 t and 3/4 t. Both figures illustrate the crossover of the limit curves imposed by the several regions and the need for composite limit curves.

Figures 5-3 and 5-4 are very similar to 5-1 and 5-2; however, they are for normal cooldown. Note that the beltline region steady-state limit curves for normal cooldown control the composite limit curves for 5 and 32 EFPY. At the high fluid temperatures (T_f > 205F) the normal cooldown composite limit curve for 5 EFPY (Figure 5-3) is less restrictive than the curve for normal heatup (Figure 5-1). This is primarily due to the large difference between the fluid temperature and the closure head region wall metal temperature at 5/6 t that occurs during heatup. However, at the lower fluid temperature (T_f < 124F), Figure 5-3 is more restrictive than Figure 5-1 because of the contributing thermal stresses at the inside corner of the outlet nozzles. The presence of the thermal stresses reduces the maximum allowable pressure. Again, Figures 5-3 and 5-4 illustrate the need for the composite limit curves.

Figures 5-5 and 5-6 present the limits imposed by the several regions of the reactor vessel and the composite limit curves for the PSHT and ISLHT. The allowable pressure-temperature combinations of these figures differ because of the different sizes of the postulated flaws, applied margins of safety, and assumed RT_{NDT}s. For both tests the limit imposed by the closure head region during heatup control the composite limit curves. For the

ISLHT the limits imposed by the outlet nozzles during cooldown also control the composite limit curve at the low temperatures. However, for the ISLHT, the beltline region would eventually control at higher EFPY. At 5 EFPY the temperature difference between the RT_{NDT}s of the beltline and the closure head region materials is not large enough to compensate for the higher stress intensities that the closure head region is subjected to (at the same internal pressure).

Figure 5-7 shows the development of the minimum pressure-temperature limit curve for reactor core operation up to 5 EFPY based on Appendix G to 10 CFR 50. The references used here are the limit curve for normal heatup and the minimum permissible temperature for the ISLHT pressure. The data used for Figure 5-7 are the composite limit curve of Figure 5-1 and the minimum permissible temperature for 2500 psi obtained from Figure 5-6. The criticality limits imposed by the Technical Specifications are based on other considerations since these limits are not controlling.

5.2. Technical Specification Limit Curves

The Technical Specificiations for each plant give allowable pressure and temperature combinations and require that the RC system be maintained within these limits during normal heatup and cooldown, criticality, and inservice leak and hydrostatic tests. The objective of these pressure-temperature limits is to prevent stresses from exceeding the ASME Code maximum allowable design stresses and the stresses allowed by ASME Appendix G for protection against nonductile failure. Since the stresses allowed by ASME Appendix G are generally more restrictive than the Code maximum allowable design stresses, the Technical Specifications pressure-temperature limits are the nonductile fracture prevention limits presented in section 5.1. However, there is one exception:

During cooldown, the stresses in the steam generator tubing may exceed the ASME Code maximum allowable stresses if cooldown rates are high, and the allowable pressure-temperature combination during cooldown is calculated according to ASME Appendix G. When high cooldown rates are desired, the pressure-temperature limit curve is modified by reducing the allowable pressure, which reduces stresses in the steam generator tubing.

Figures 5-8 through 5-10 are pressure-temperature limit that illustrate the limit curves that appear in the Technical Specifications of a typical 177 FA type plant. Figure 5-8 represents the normal heatup limits applicable for the first 5 EFPY. Figure 5-9 represents the normal cooldown limitations, and Figure 5-10 represents the ISLHT limits. These figures were obtained from Figures 5-1, 5-3, and 5-6, respectively. Figure 5-7 was also used to develop Figure 5-8. Figures 5-1, 5-3, and 5-8 were adjusted as follows:

1. The maximum allowable pressure had been reduced by the pressure differential between the point of system pressure measurement and the limiting region of the reactor vessel for all operating pump combinations. The applied pressure differential is 100 psig when either the beltline region or the outlet nozzles control the pressure-temperature curves and 75 psig when the closure head region controls them. These pressure differentials have been conservatively calculated.

2. Figures 5-3 and 5-6 were adjusted to include the pressure-temperature limits imposed by the steam generator tubing.

For some 177 FA plants and other B&W plants, the Technical Specification limit curves will be different from those presented in Figures 5-8 through 5-10. The differences are caused by the lower maximum allowable ramp rates and the material's RT_{NDT}, wall thickness, neutron fluence, etc.

5.3. Preservice System Hydrostatic Test Limit Curve

The Technical Specifications do not include the RC system pressure-temperature limits for the PSHT since this test is conducted before the plant operating license is issued. The limits for the PSHT are imposed by the test procedure.

Figure 5-11 is the PSHT limit curves as developed by adjusting the composite limit curve of Figure 5-5. The adjustments are the same as those used to develop Figures 5-8 through 5-10. Figure 5-11 is labeled as the PSHT limit curve for the typical 177-FA type plant. The actual curves may differ since this curve was calculated using 100°/hour heatup and cooldown ramp rates and during the PSHT, the ramp rates are much lower than 100F/hour.

Table 5-1. Unirradiated Impact Properties and Residual Element Content of Beltline Region Materials in a Typical 177 FA Plant

Material identification type, location	Core MP to weld CL, in.	Transverse C_V USE, ft-lb	RT_{NDT}, F	Chemistry, %		
				Cu	P	S
A. SA 508 Class 2, nozzle belt	--	183	+10	0.054	0.008	0.006
B. SA 533 B upper shell	--	88	+20	0.20	0.008	0.016
C. SA 533 B upper shell	--	90	+20	0.20	0.008	0.016
D. SA 533 B lower shell	--	119	-20	0.12	0.013	0.015
E. SA 533 B lower shell	--	99	+40	0.12	0.013	0.015
F. Weld, upper long.	--	(66)[a]	(+20)	0.20	0.009	0.009
G. Weld, upper circ	123	(66)	(+20)	0.19	0.021	0.016
H. Weld, mid circ (100%)	-63	(66)	(+20)	0.27	0.014	0.011
I. Weld, lower long. (100%)	--	(66)	(+20)	0.22	0.015	0.013
J. Weld, lower circ (100%)	-249	(66)	(+20)	0.20	0.015	0.021
K. Weld, outlet nozzle	244.8	(66)	(+20)	0.19	0.021	0.051

[a]Numbers in parentheses indicate predicted values.

Table 5-2. Typical Material Data for Preparing Beltline Region Pressure-Temperature Limit Curves

Mat ID	Initial RT_{NDT}, F	End of 5th EFPY: Fluence, $E \geq 1$ MeV n/cm^2		End of 5th EFPY: RT_{NDT}, F		End of 5th EFPY: RT_{NDT}, F		End of 32nd EFPY: Fluence, $E \geq 1$ MeV n/cm^2		End of 32nd EFPY: RT_{NDT}, F		End of 32nd EFPY: RT_{NDT}, F	
		1/4 t	3/4 t	1/4 t	3/4 t	1/4 t	3/4 t	1/4 t	3/4 t	1/4 t	3/4 t	1/4 t	3/4 t
A	+10	2.63E18	5.9E17	30	16	40	26	1.68E19	3.8E18	105	35	115	45
B	+30	2.63E18	5.9E17	75	38	105	68	1.68E19	3.8E18	220	90	260	120
C	+20	2.63E18	5.9E17	76	38	95	58	1.68E19	3.8E18	220	90	240	110
D	-20	2.63E18	5.9E17	45	21	25	1	1.58E19	3.8E18	140	59	130	39
E	+40	2.63E18	5.9E17	45	21	85	61	1.68E19	3.8E18	150	69	180	99
F	+20	2.25E18	5.1E17	70	34	90	54	1.44E19	3.26E18	212	85	232	[illegible]
G	+20	2.63E18	5.9E17	75	38	95	58	1.68E19	3.8E18	220	90	240	[illegible]
H	+20	2.63E18	5.9E17	75	38	95	58	1.68E19	3.8E18	220	90	240	[illegible]
I	+20	2.0E18	4.53E17	64	32	84	52	1.28E19	2.9E18	200	80	220	100
J	+20	<8.75E16	--	0	0	20	20	<5.6E18	--	<50	--	<70	--
K	+20	<8.75E16	--	0	0	20	20	<5.6E18	--	<50	--	<70	--

Table 5-1. Unirradiated Impact Properties and Residual Element Content of Beltline Region Materials in a Typical 177 FA Plant

Material identification type, location	Core MP to weld CL, in.	Transverse C_v USE, ft-lb	RT_{NDT}, F	Chemistry, %		
				Cu	P	S
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C. SA 533 B upper shell	--	90	+20	0.20	0.008	0.016
D. SA 533 B lower shell	--	119	-20	0.12	0.013	0.015
E. SA 533 B lower shell	--	99	+40	0.12	0.013	0.015
F. Weld, upper long.	--	(66)[a]	(+20)	0.20	0.009	0.009
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H. Weld, mid circ (100%)	-63	(66)	(+20)	0.27	0.014	0.011
I. Weld, lower long. (100%)	--	(66)	(+20)	0.22	0.015	0.013
J. Weld, lower circ (100%)	-249	(66)	(+20)	0.20	0.015	0.021
K. Weld, outlet nozzle	244.8	(66)	(+20)	0.19	0.021	0.061

[a]Numbers in parentheses indicate predicted values.

Table 5-2. Typical Material Data for Preparing Beltline Region Pressure-Temperature Limit Curves

		End of 5th EFPY						End of 32nd EFPY					
		Fluence, $E \geq 1$ MeV n/cm^2		RT_{NDT}, F		RT_{NDT}, F		Fluence, $E \geq 1$ MeV n/cm^2		RT_{NDT}, F		RT_{NDT}, F	
Mat ID	Initial RT_{NDT}, F	1/4 t	3/4 t	1/4 t	3/4 t	1/4 t	3/4 t	1/4 t	3/4 t	1/4 t	3/4 t	1/4 t	3/4 t
A	+10	2.63E18	5.9E17	30	16	40	26	1.68E19	3.8E18	105	35	115	45
B	+30	2.63E18	5.9E17	75	38	105	68	1.68E19	3.8E18	220	90	260	120
C	+20	2.63E18	5.9E17	76	38	95	58	1.68E19	3.8E18	220	90	240	110
D	-20	2.63E18	5.9E17	45	21	25	1	1.58E19	3.8E18	140	59	130	39
E	+40	2.63E18	5.9E17	45	21	85	61	1.68E19	3.8E18	150	69	180	99
F	+20	2.25E18	5.1E17	70	34	90	54	1.44E19	3.26E18	212	85	232	110
G	+20	2.63E18	5.9E17	75	38	95	58	1.68E19	3.8E18	220	90	240	110
H	+20	2.63E18	5.9E17	75	38	95	58	1.68E19	3.8E18	220	90	240	110
I	+20	2.0E18	4.53E17	64	32	84	52	1.28E19	2.9E18	200	80	220	100
J	+20	<8.75E16	--	0	0	20	20	<5.6E18	--	<50	--	<70	--
K	+20	<8.75E16	--	0	0	20	20	<5.6E18	--	<50	--	<70	--

Figure 5-1. Normal Heatup Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 5 EFPY

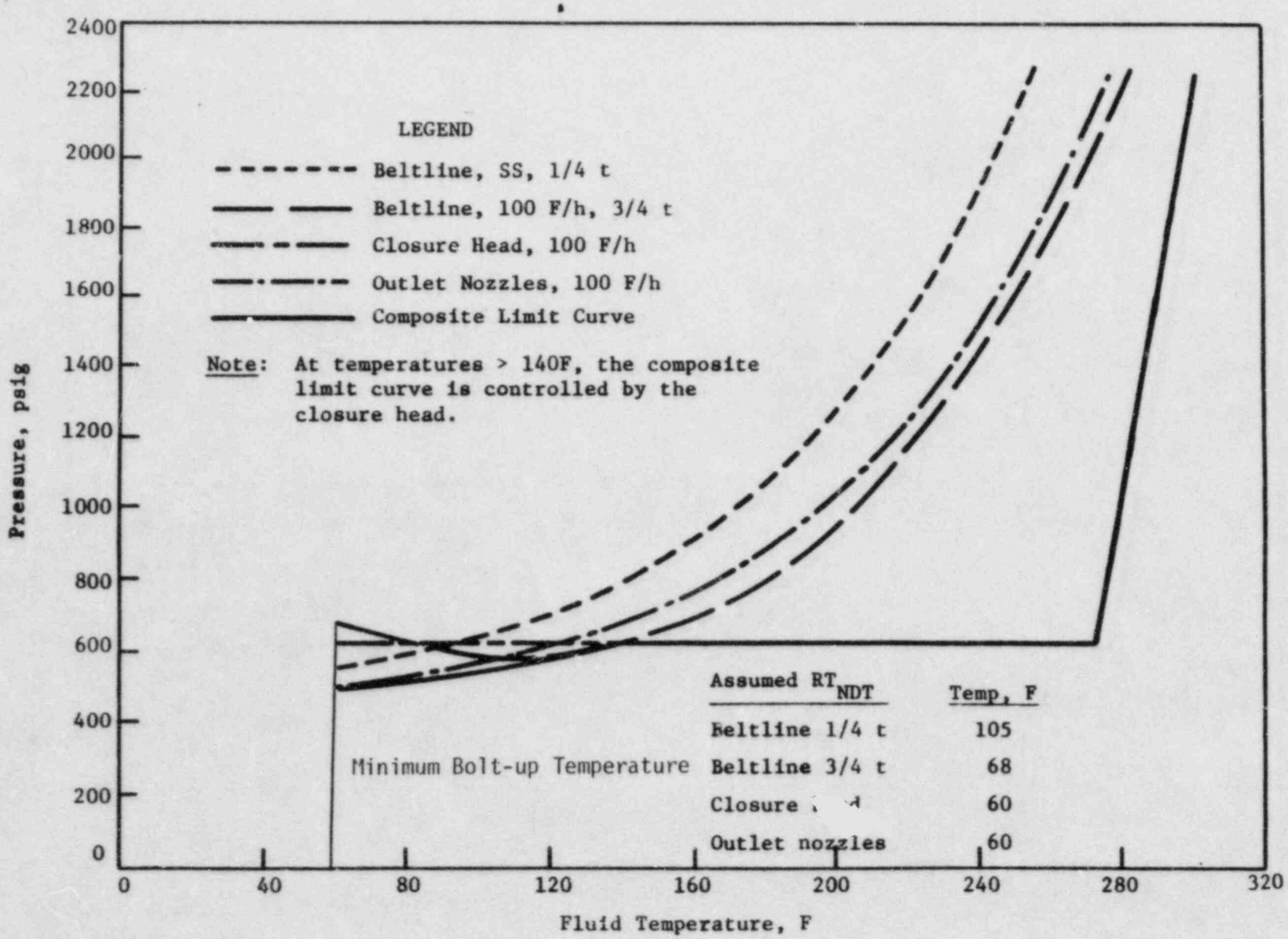

Figure 5-2. Normal Heatup Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 32 EFPY

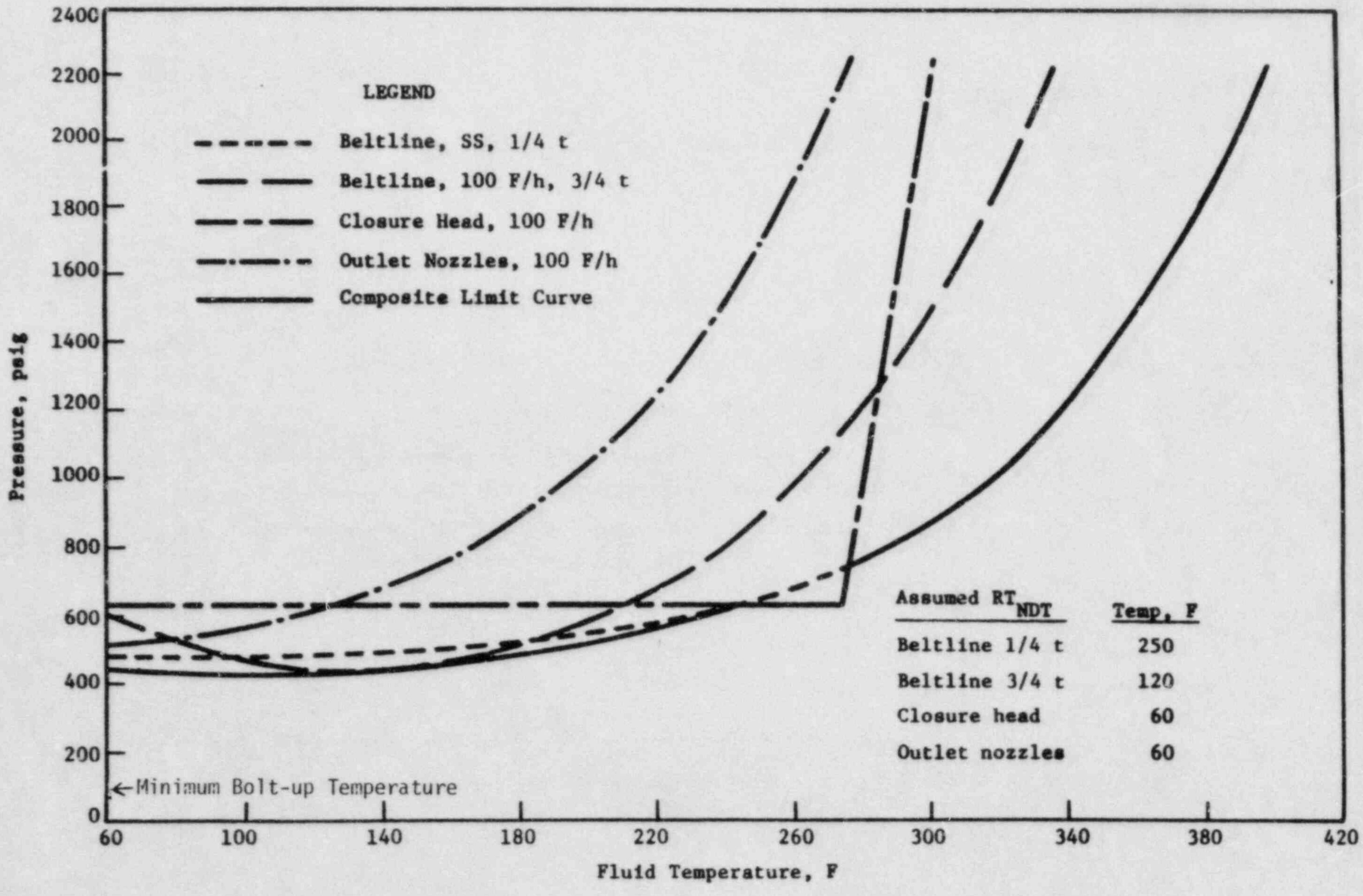

Figure 5-3. Normal Cooldown Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 5 EFPY

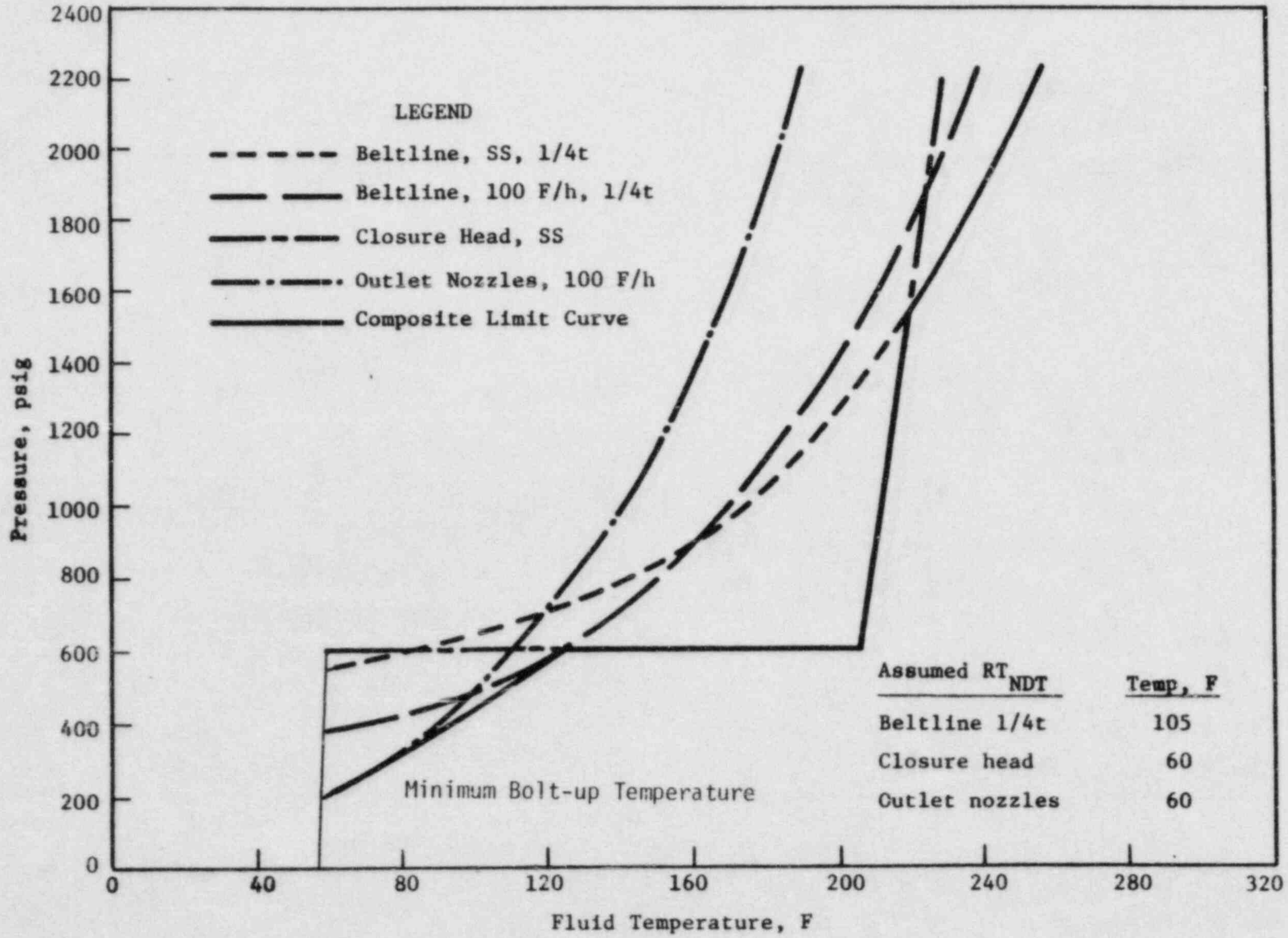

Figure 5-4. Normal Cooldown Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 32 EFPY

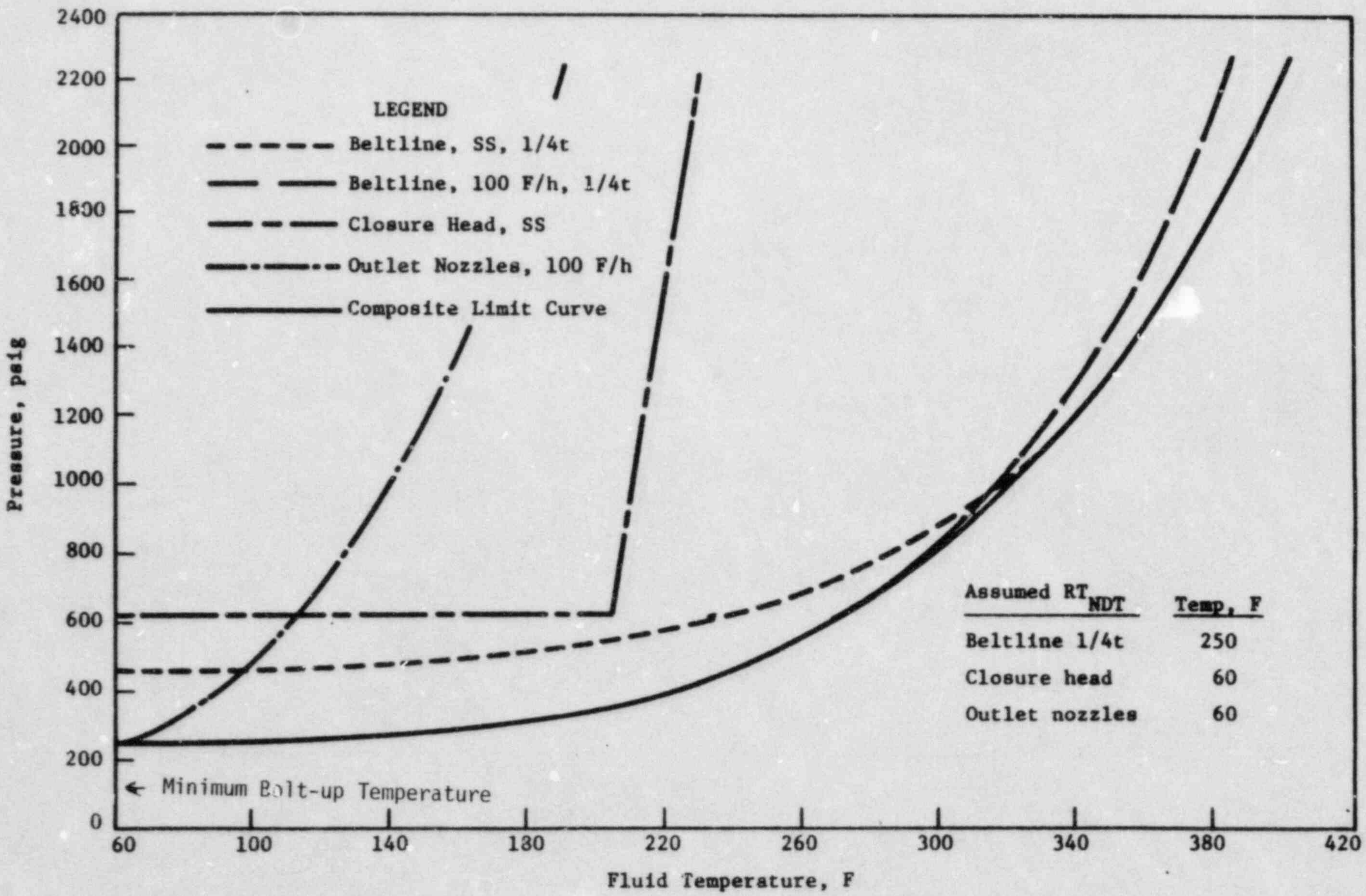

Figure 5-5. PSHT Heatup and Cooldown Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve

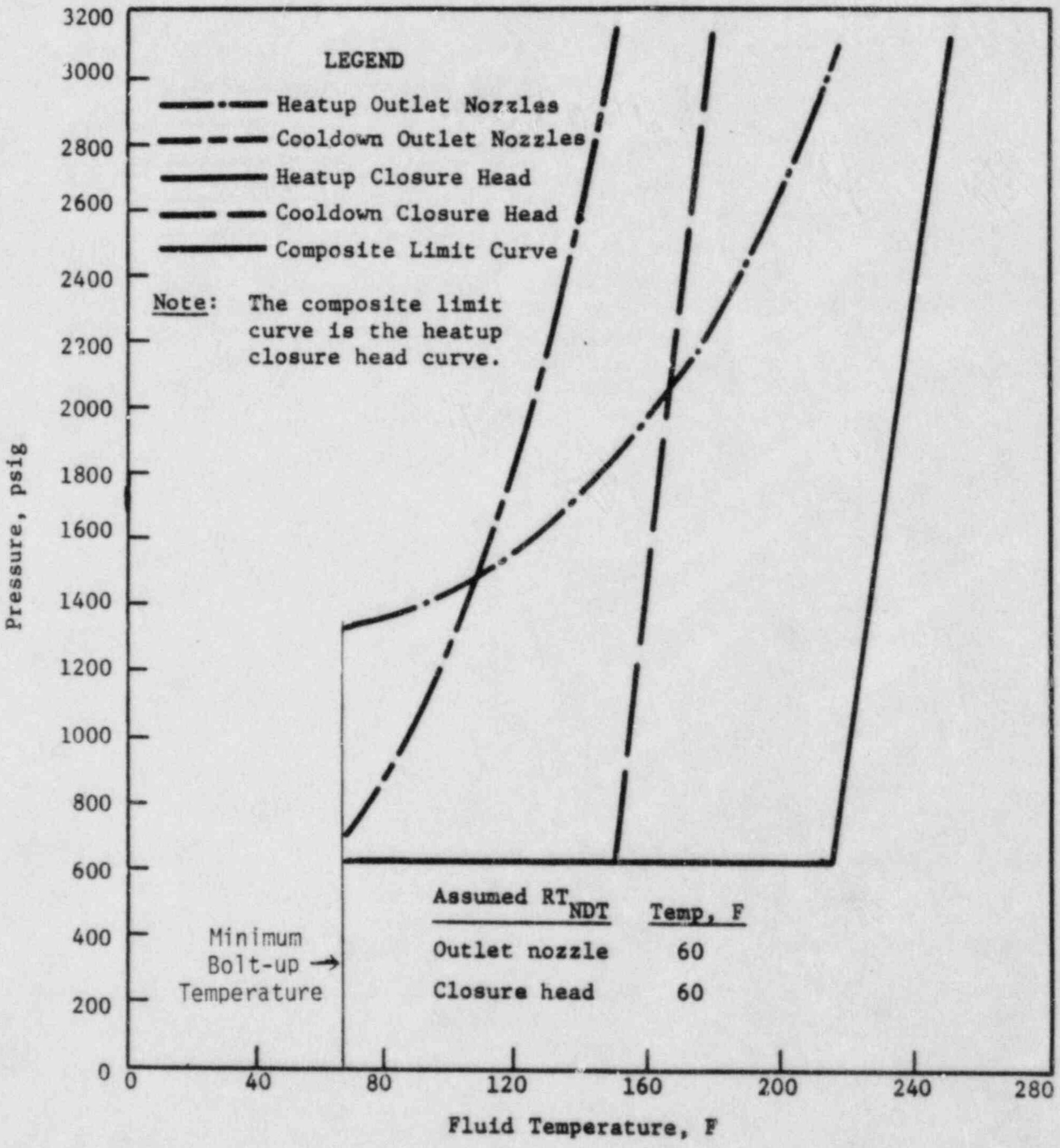

Figure 5-6. ISLT Heatup and Cooldown Pressure-Temperature Limits Imposed by Several Reactor Vessel Regions and Composite Limit Curve for 5 EFPY

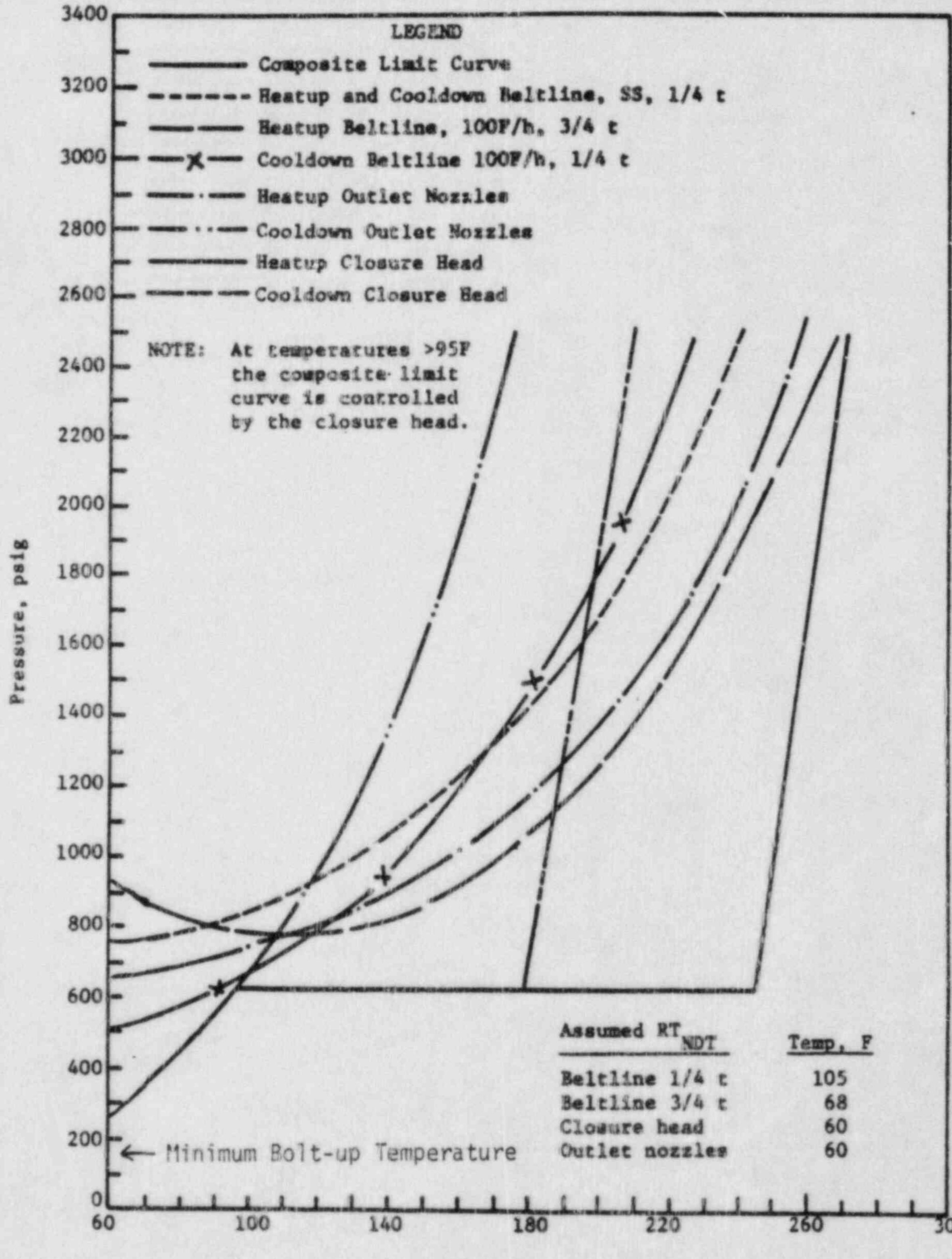

Figure 5-7. Determination of Reactor Core Operation Pressure-Temperature Curve for 5 EFPY per Appendix G to 10 CFR 50

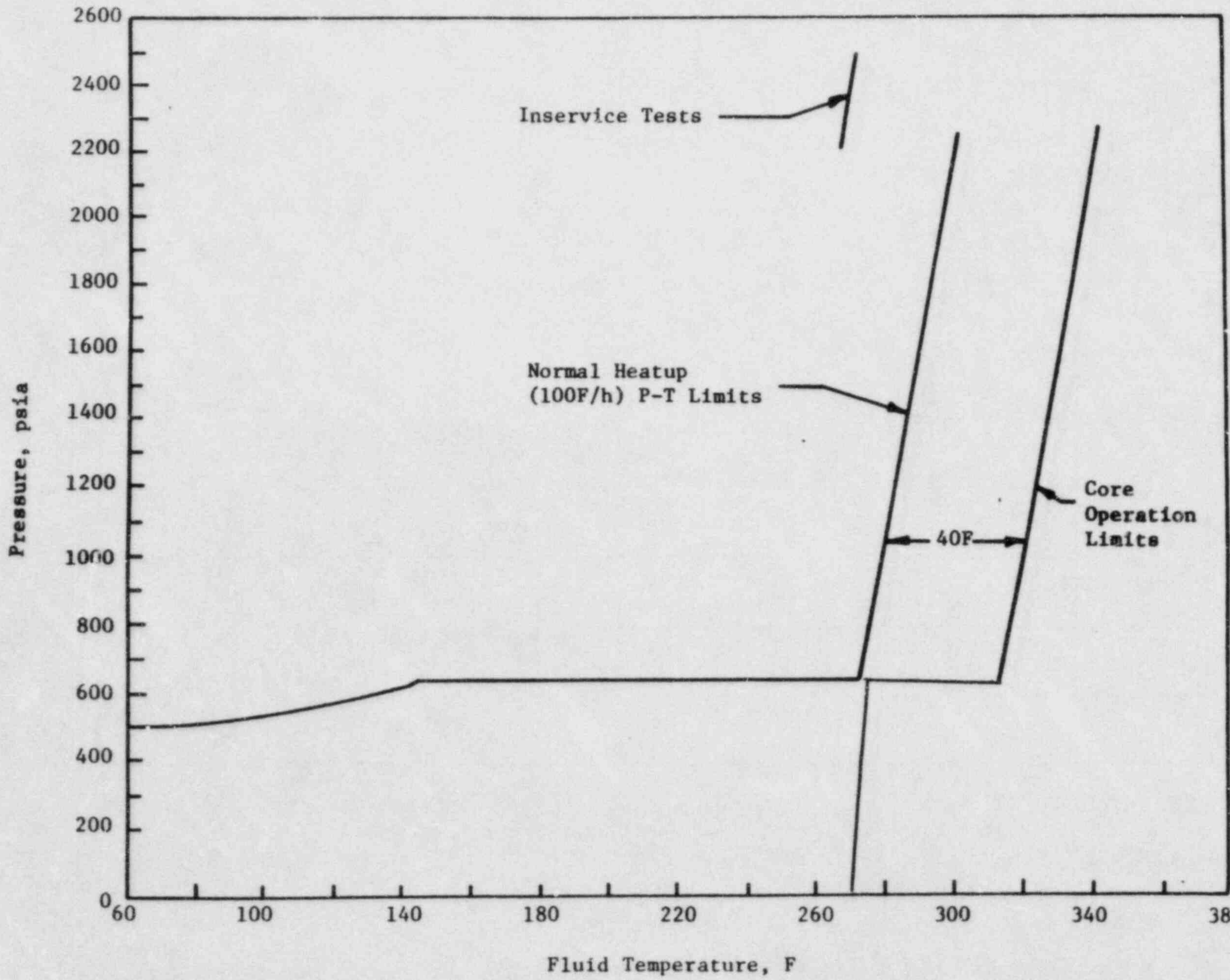

Figure 5-8. Normal Operation Heatup Pressure-Temperature Limit Curves for Typical Plant Technical Specifications, Applicable up to 5 EFPY

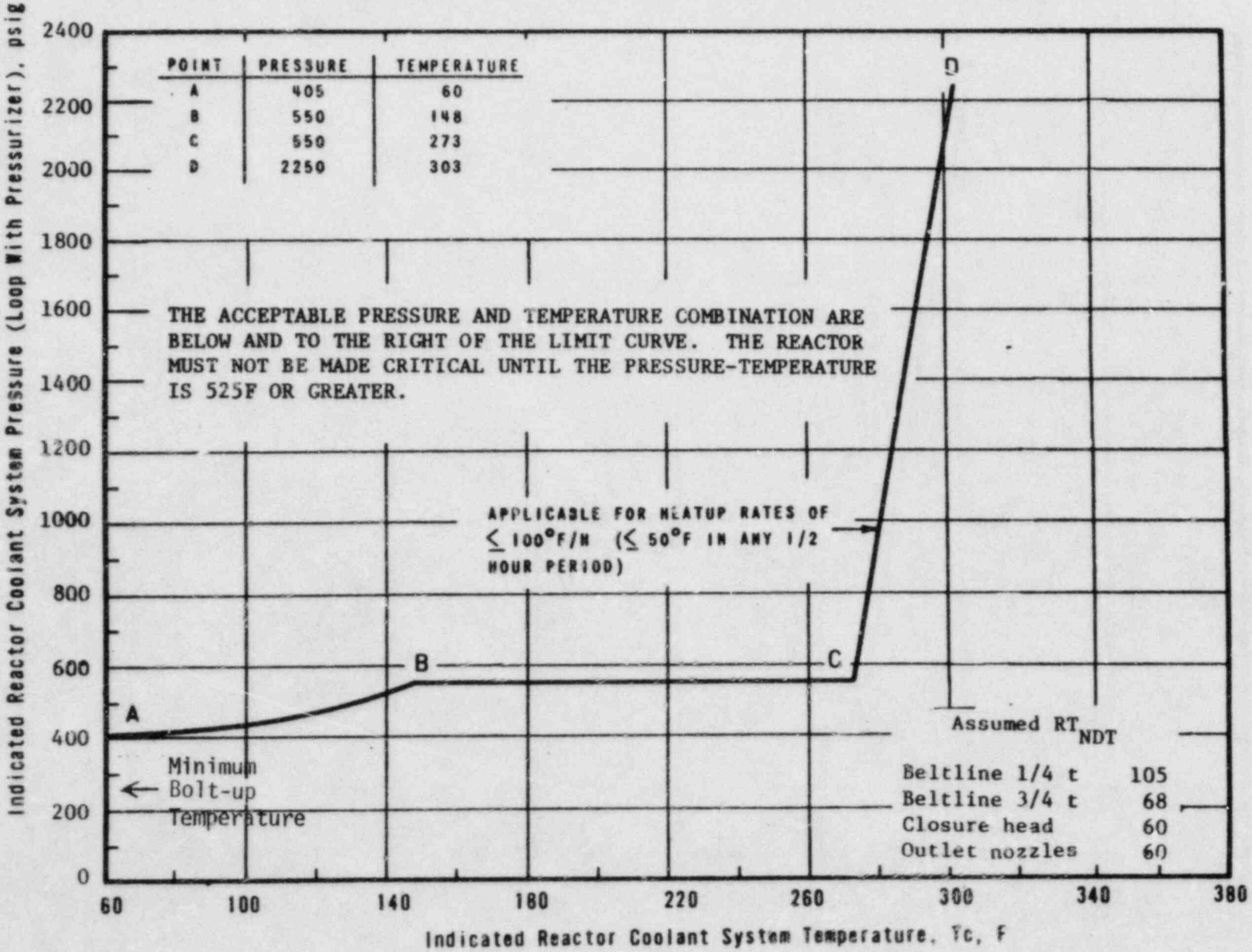

Figure 5-9. Normal Operation Cooldown Pressure-Temperature Limit Curve for Typical Plant Technical Specification, Applicable up to 5 EFPY

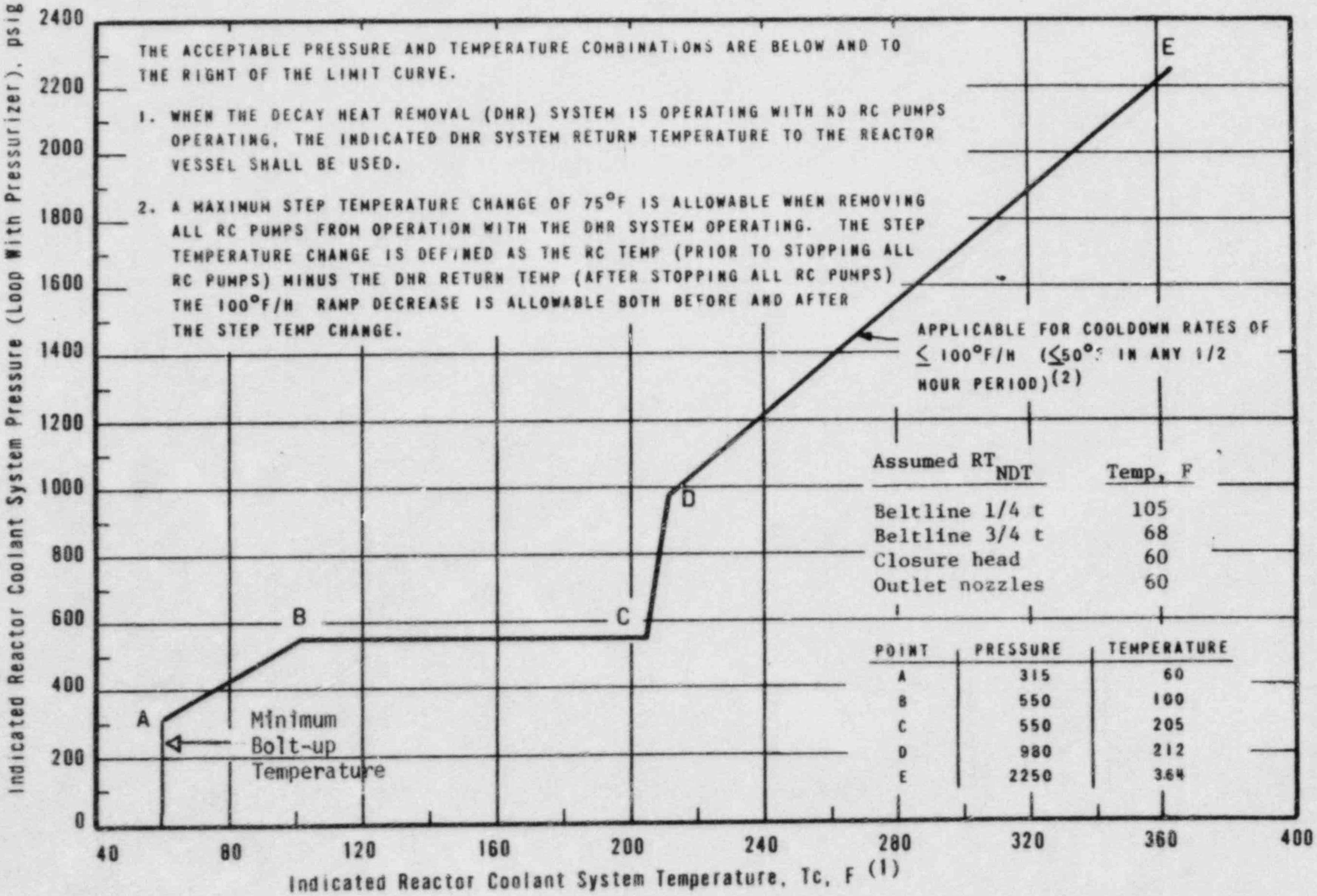

Figure 5-10. Inservice Leak and Hydrostatic Test Heatup and Cooldown Pressure-Temperature Limit Curve for Typical Plant Technical Specifications, Applicable up to 5 EFPY

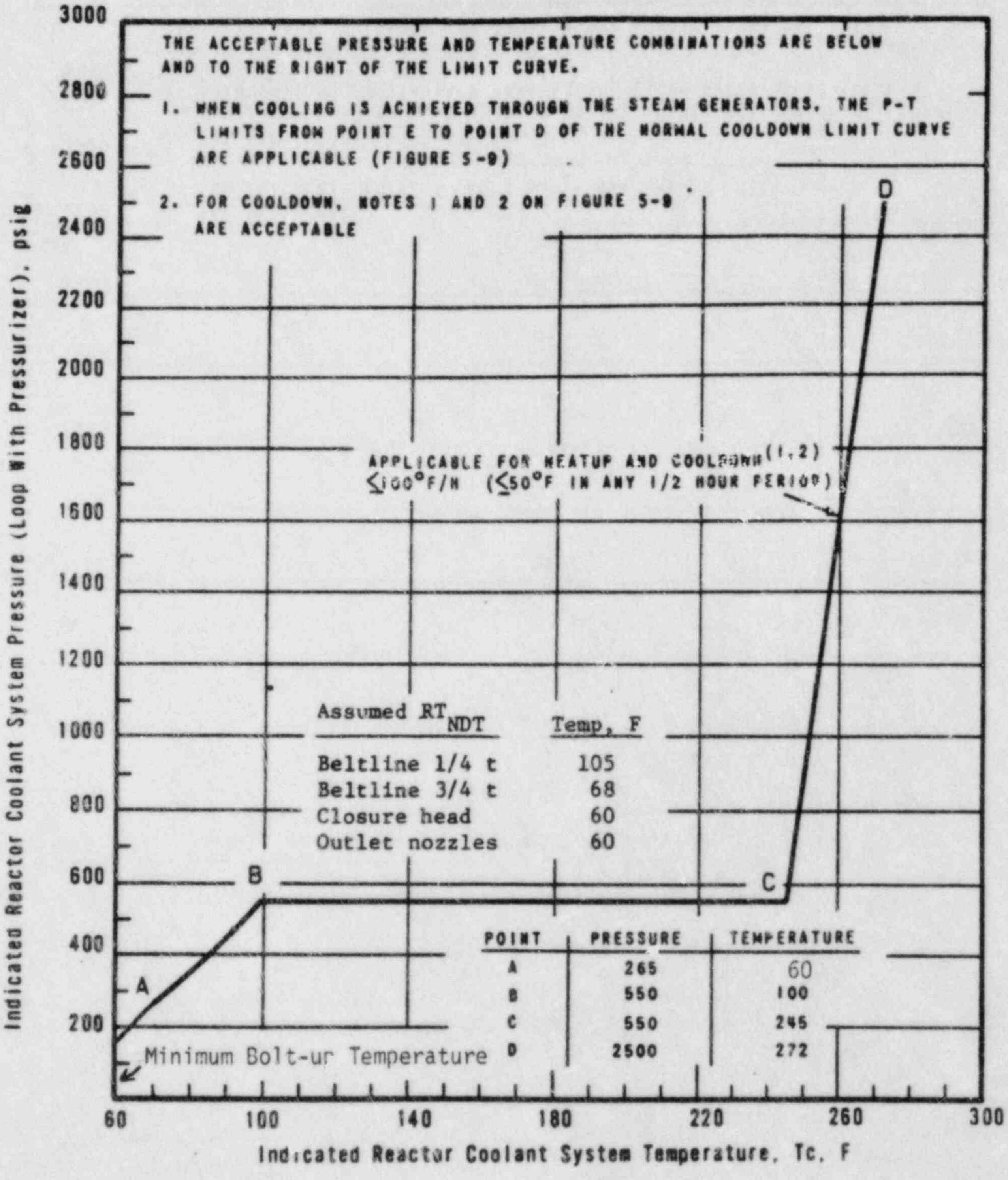

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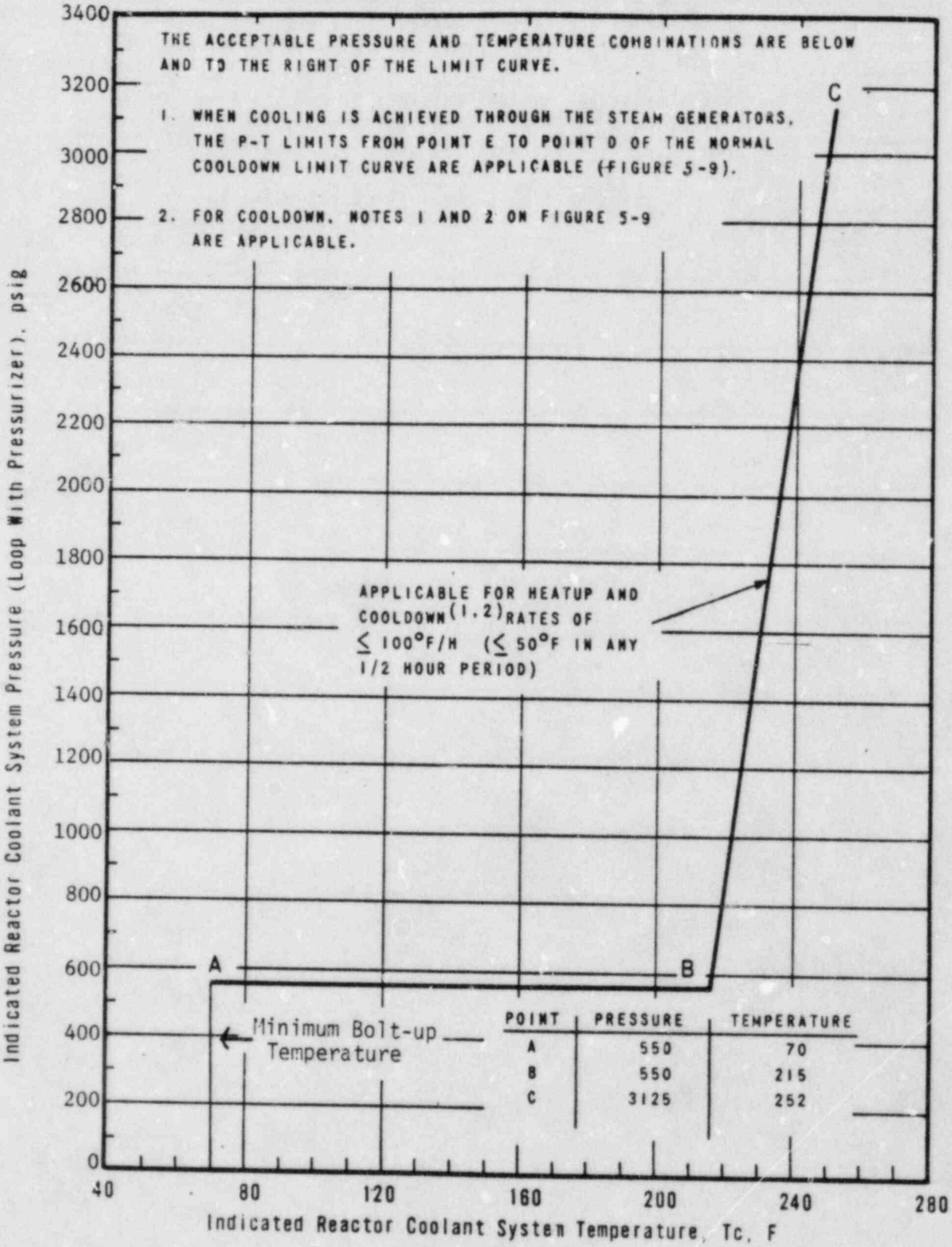
Figure 5-11. PSHT Pressure-Temperature Limit Curve for Typical Plant



POINT	PRESSURE	TEMPERATURE
A	550	70
B	550	215
C	3125	252

6. EPFM ANALYTICAL PROCEDURES

6.1. Basis

The analytical procedures given in Section 4 are applicable for the areas of the pressure boundary which comply with the material restrictions of ASME Appendix G. If the material does not comply with the restrictions then supplemental analysis is required to assure the reactor coolant pressure boundary integrity. The only anticipated divergence from the materials restrictions is the failure of irradiated materials, in particular weld metal, to exhibit a Charpy upper shelf exceeding 50 ft-lbs.

If a material exhibits less than 50 ft-lbs absorbed energy but greater than 30 ft-lbs the adjusted shift in RT_{NDT} is determined in accordance with 10 CFR 50 Appendix G. The analysis of section 4 is carried out in the same manner previously discussed. 10 CFR 50 Appendix G further specifies that Charpy upper shelves below 50 ft-lbs are permitted if the component is verified to still have a margin of safety equivalent to that specified in ASME Appendix G. The only area of the reactor coolant boundary which is predicted to potentially fall below 50 ft-lbs is the reactor vessel beltline. This evaluation will be restricted to that area but similar evaluation could be performed on other areas.

Appendix G of the ASME Code is a design guideline for the preventation of non-ductile failure. The general philosophy is to index the fracture toughness to temperature and require that the component be operated at a sufficiently high temperature to preclude non-ductile failure. ASME Appendix G is not adequate to control operating conditions in the higher temperature regime. In the high temperature regime ductile tearing is the controlling mechanism for possible loss of vessel integrity. Evaluation for ductile tearing can be accomplished utilizing the J-integral and the J_{IR} curve for the material.

6.2. Elastic-Plastic Fracture Mechanics (EPFM) Analytical Model

An elastic-plastic fracture mechanics (EPFM) procedure based on deformation plasticity J-integral solutions in the format of a failure assessment diagram will be used to set the pressure-temperature limits for upper shelf material behavior.

The procedure for setting these pressure-temperature limits consists of four steps:

1. J-integral formulation.
2. Failure assessment diagram curve generation.
3. Assessment point evaluation.
4. Instability pressure prediction.

For reactor vessel materials which can be modeled by deformation plasticity and whose stress-strain behavior can be represented by a power law strain-hardening equation, the J-integral response ($J_{applied}$) can be evaluated for the reference flaw using the expression

$$J = J^e(a_{eff},P) + J^p(a,P,n) \qquad (1)$$

where J^e is the elastic contribution based on Irwin effective crack depth, a_{eff}, and J_p is the deformation plasticity contribution derived in reference 6. P is the applied pressure and n is the strain-hardening exponent. A convenient way to use this equation is to construct a deformation plasticity failure assessment diagram (DPFAD). The details of this procedure are found in references 10 and 11. The process summarized here only for the beltline flaw evaluation.

6.2.1. DPFAD Curve Generation

The DPFAD curve expression is obtained by normalizing the sum of the elastic and plastic response by the "elastic" J-integral of the flawed reactor vessel in terms of "a," where

$$J^e(a) = \frac{(1 - \nu^2)K_I^2(a)}{E} \qquad (2)$$

and K_I is the linear-elastic fracture mechanics (LEFM) stress intensity factor. E and ν are Young's modulus and Poisson's ratio, respectively. The normalized J-response is then defined by

$$K_r = \sqrt{\frac{J^e}{J}} = f(S_r) \tag{3}$$

where $S_r = P/P_L(a)$.

P is the applied pressure and P_L is the reference plastic collapse pressure or limit pressure, a function of "a" and the material yield strength, σ_0.

Equation 3 defines a DPFAD curve which is a function of the flaw geometry, structural configuration, and stress-strain behavior of the material of interest. This curve, in terms of K_r, S_r is independent of the magnitude of the applied loading.

For the beltline area of the reactor vessel assuming a semi-elliptical axial flaw on the inside of the vessel.[9]

$$K_I = \frac{PR_i}{t} \sqrt{\frac{\pi a}{Q}} \quad F(a/\ell, a/t) \tag{4}$$

where
- P = applied pressure,
- R_i = inside radius,
- t = thickness,
- $F = .97[M_1 + M_2(a/t)^2 + M_3\ (a/t)^4]fc$,
- $M_1 = 1.13 - 0.18\ a/\ell$,
- $M_2 = -.54 + .445/\ (.1 + a/\ell)$,
- $M_3 = .5 - 1/(.65 + 2a/\ell) + 14\ (1-2a/\ell)^{24}$,
- $fc = 1.152 - .05\ \sqrt{a/t}$.
- $Q = 1 + 4.593(a/\ell)^{1.65}$
- ℓ = length of flaw
- a = flaw depth

then

$$J^e(a) = \frac{P^2 R_i^2}{t^2} \ \frac{\pi a}{Q} \frac{F^2\ (1 - \nu^2)}{E} \tag{5}$$

The effective crack correction is given by

$$a_{eff} = a + \frac{1}{6\pi} \frac{(n-1)}{(n+1)} \frac{K_I^2}{\sigma_o^2} \phi; \ \phi = \frac{1}{1 + S_r^2}$$

where
- n = strain hardening exponent, (Ramberg-Osgood)
- σ_0 = engineering yield stress,
- $S_r = P/P_L$,

$$P_L = \frac{2}{\sqrt{3}} \sigma_o \frac{(t - a^*)}{(R_i + a^*)}$$

The limit pressure expression P_L is based on a continuous axial flaw. A correction is applied in the form of a* to account for the partial length flaw.

$$a^* = \frac{a(1 - s)}{1 - (a/t)s} \tag{6}$$

where:

$$s = (1 + \ell^2/2t^2)^{-1/2}$$

The plastic portion of J is given by the following expression

$$J^P = \frac{\alpha\sigma_o^2}{E} a(1 - a/t)h_1(P/P_L)^{n+1} \tag{7}$$

where α is obtained from the Ramberg-Osgood stress-strain relation and h_1 is a dimensionless term which is a function of a/t, a/ℓ, n and t/R_i. This latter constant is evaluated from finite element results.[8]

Combining all of the above terms into equation (3) results in an equation which when plotted has the shape shown in Figure 6-1. The DPFAD curve is unique for a given set of stress-strain parameters, flaw size and vessel geometry.

6.2.3. Assessment Point Evaluation

Having defined the DPFAD curve the beltline of the vessel can be evaluated for a given set of material properties. Assessment points are denoted by K_r', S_r' and are defined as follows:

$$K_r'(a_o + \Delta a) = \sqrt{\frac{J^e(a_o + \Delta a)}{J_R\ (\Delta a)}} \tag{8}$$

$$S_r'(a_o + \Delta a) = \frac{P}{P_L\ (a_o + \Delta a)} \tag{9}$$

where terms are as defined in section 6.2.1 with $J_R(\Delta a)$ being the material J_I-R resistance property. a_o is the initial assumed flaw.

6.2.4. Instability Pressure Prediction

To evaluate the structure, the applied pressure is held constant and successive points are calculated incrementing the crack size. The assessment point which is the minimum distance from the origin represents the maximum crack growth which the structure can sustain before becoming unstable. The corresponding point on the DPFAD curve then represents the instability pressure designated by P_{crit}. This process is illustrated in Figure 6-2.

If the initial point evaluated is $J_R(\ a) = J_{IC}$ then the pressure can be determined which corresponds to the initiation of ductile tearing. This pressure is designated P_{init} and is also illustrated on Figure 6.2.

6.2.5. Sample Calculation and Presentation of Data

For further clarification of the failure assessment diagram approach to predicting tearing pressure, a sample calculcation is presented of an ASME Section III, Appendix G flaw in a beltline weld in a reactor pressure vessel under a pressure of 2500 psi.

Figure 6-3 and Table 6-1 present the FAD format while Figure 6-4 presents a plot of the J_R (Δa) curve for the weld material. Figure 6-5 shows the resultant tearing pressure versus stable crack growth, Δa, as well as the local 2500/S_r' (S_r' is given in Table 6-12 as a function of Δa). The critical pressure is the lower value of the two curves. In all the figures and the table, the numbers refer to the points plotted; initiation is the point numbered #1 ($J=J_{IC}$) while the instability or the critical point is numbered #5.

6.4. Thermal Stress

Thermal stresses are not considered when evaluating ductile tearing. Thermal stresses arising from radial gradients through the wall are self-limiting and will decrease with crack propagation. Furthermore, for the conditions being considered (i.e., normal and upset transients in the power operating range) the thermal contribution to the J applied is small calculated on an elastic basis.

6.5. Acceptance Criteria

The acceptance criteria for the evaluation are two fold. Although the flaw used in the evaluation is hypothetical it is necessary to demonstrate that ductile initiation will not occur-to preclude assuming incremental reference flaws throughout the life of the plant. Therefore, the first criteria is that the initiation pressure, P_{init}, must be greater than 3000 psi. This value is one-third above the nominal operating pressure and ten-percent above any normal or upset anticipated transients.

The second criteria is that the instability pressure, Pcrit, must exceed two times the nominal operating pressure. For B&W designed nuclear vessels this corresponds to 4500 psi. These criteria will be reflected in the Owner's licensing document by specifying a maximum allowed pressure in the Technical Specification of 2750 psi for temperatures in the operating range.

Figure 6-1. Typical DPFAD Curve

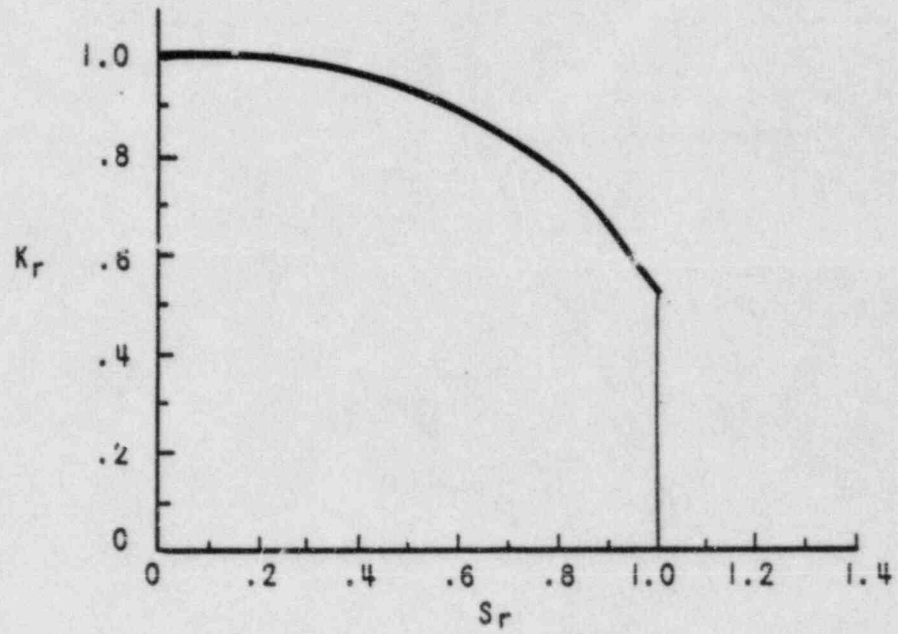


Figure 6-2. Assessment Point Illustration

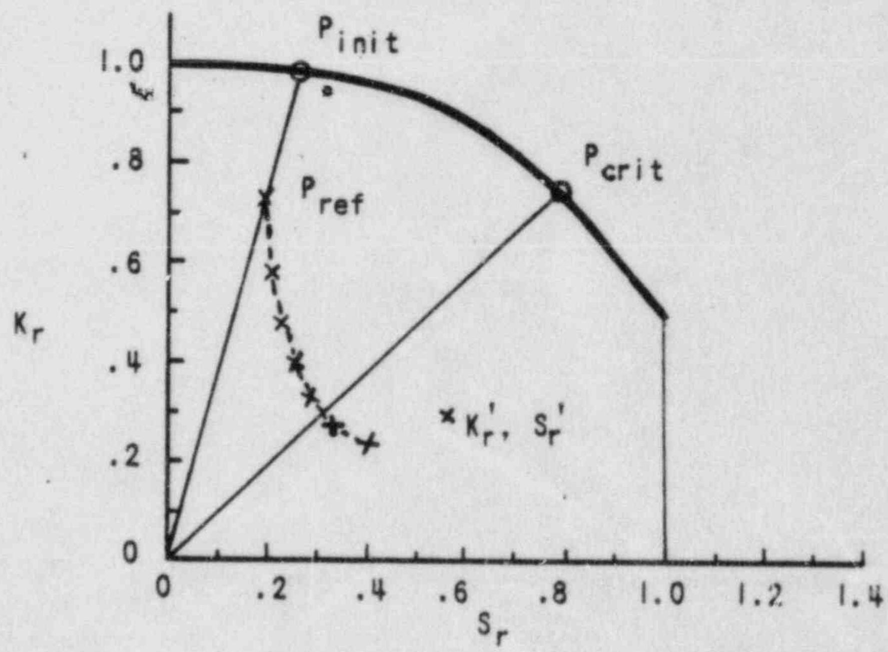
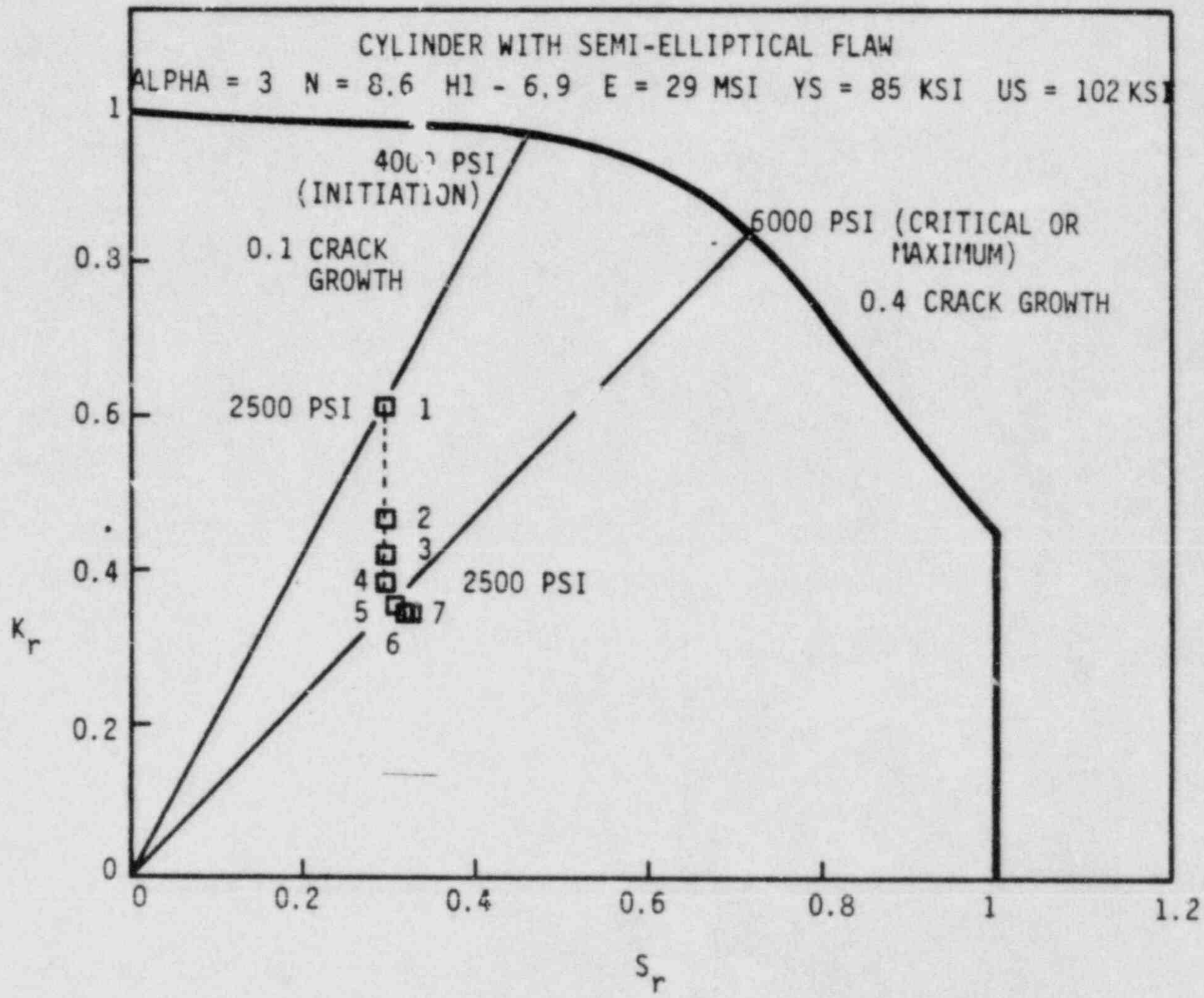


Figure 6-3. Failure Assessment Diagram Procedure Applied to Typical Beltline Region Weld



PT	a (inches)	J_r (in.-lb/in.^{2})	S_r'	K_r'	Safety Factor (S. F)
1	0.010	456 (J_{Ic})	0.284	0.611	1.60
2	0.050	818	0.286	0.465	2.03
3	0.100	1045	0.287	0.418	2.20
4	0.200	1334	0.291	0.380	2.33
5	0.400	1702 (J_{crit})	0.299	0.354	2.40
6	0.600	1963	0.308	0.346	2.39
7	0.800	2173	0.318	0.345	2.35

Tearing Pressure = 2500 psi x S. F.

Figure 6-4. Typical Beltline Weld J_R Curve

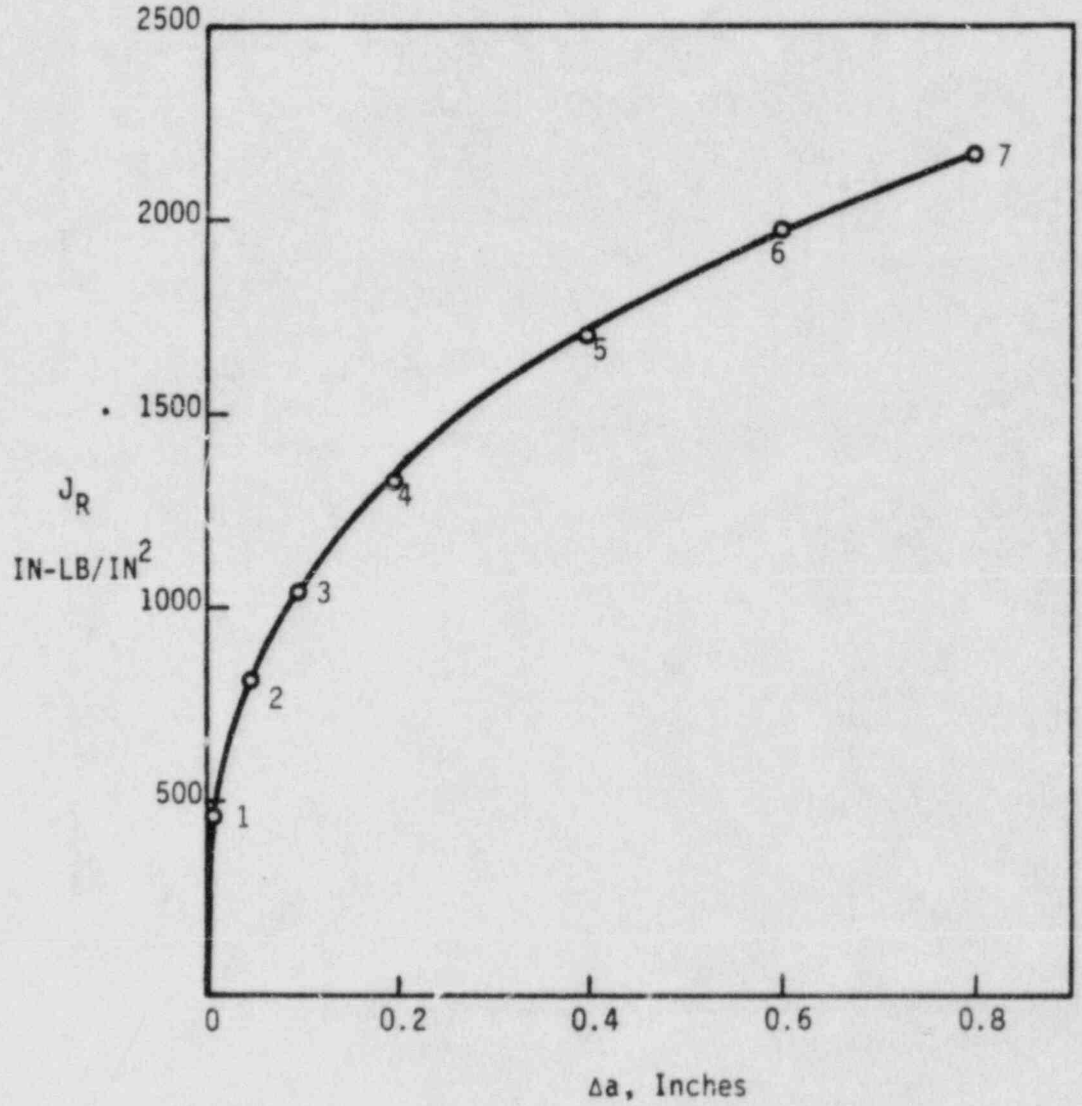
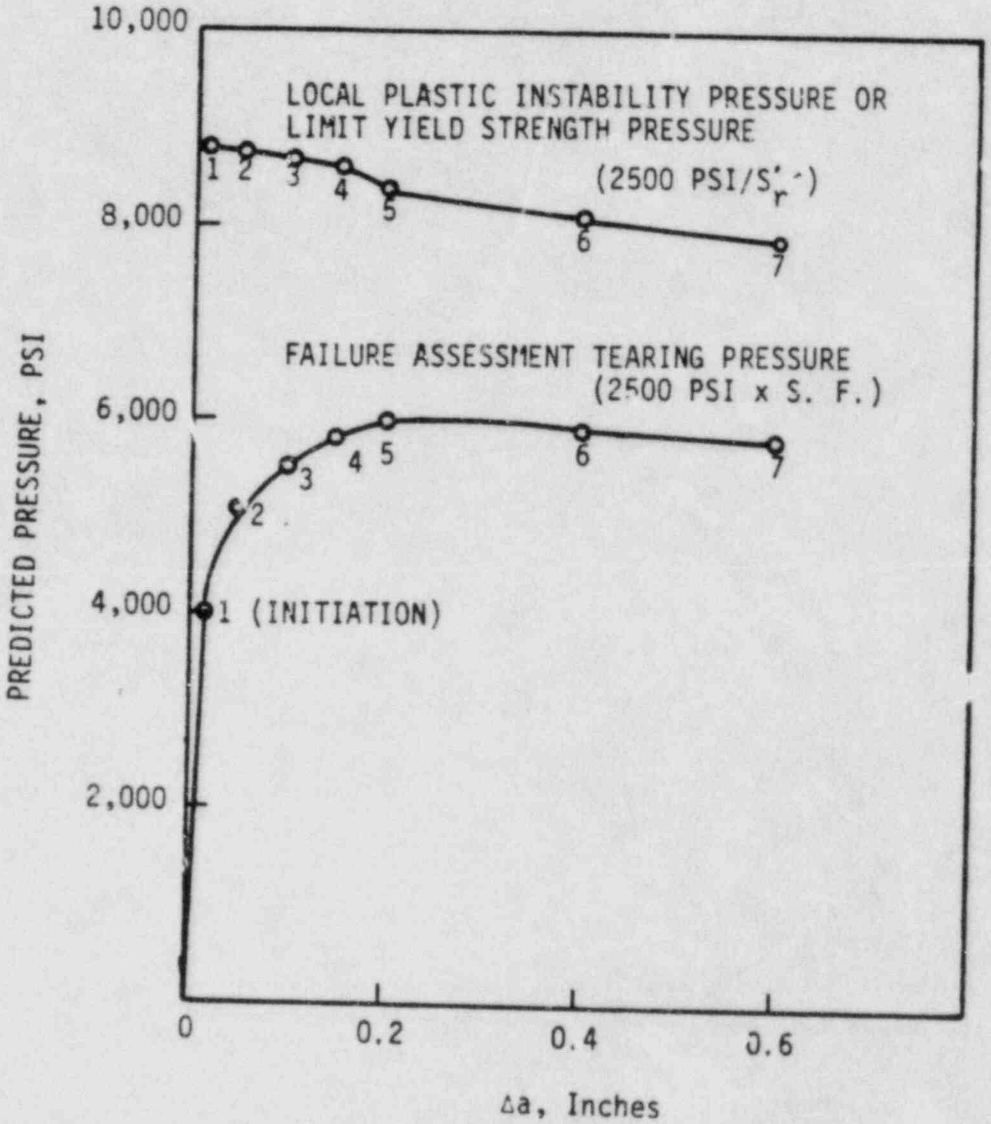


Figure 6-5. Typical Resultant Tearing Pressure Prediction



7. SUMMARY AND CONCLUSIONS

B&W's methods of compliance with the material properties and operational limit requirements of Appendix G to 10 CFR 50 have been described. Since Appendix G specifies fracture toughness requirements for the ferritic materials used in the RCPB and provides guidelines for determining its operating limitations, the RCPB is described first.

Section 2.3 describes the operational parameters of each loading condition for which pressure-temperature limit curves are required; these conditions are as follows:

1. Normal operation, including bolt preloading, heatup, and cooldown.
2. Preservice system hydrostatic test.
3. Inservice system leak and hydrostatic test.
4. Reactor core operations.

Section 3 describes the methods of compliance with these material requirements. Section 3.1 covers ferritic materials other than bolting and type 403 stainless steels. As required by Appendix G to 10 CFR 50, the RT_{NDT}s of these materials must be established in order to determine the pressure-temperature limit curves for the RC system. For later plants (ordered according to the Summer 1972 Addenda to ASME Section III or subsequent editions or addenda), the RT_{NDT}s were obtained as required by the applicable ASME Code. The RT_{NDT}s of the other ferritic materials in the older plants were conservatively estimated using the fracture toughness data obtained on low-alloy steel forgings, plates, carbon steel plates, weld metals, HAZs and piping.

Appendix G (10 CFR 50) also requires full Charpy test curves on the beltline region materials to determine the USEs for the more recent reactor vessels. B&W has specified complete Charpy test curves (normal and parallel to the principal working direction) on the base metals; for weld metals, only one curve is needed. These curves were also obtained for the HAZs of the beltline region base metals(s) selected to be monitored by the reactor vessel surveillance program. For older reactor vessels, Charpy test curves (both directions) were obtained on the materials from the surveillance program. Where enough material was available for testing, Charpy test curves for specimens oriented normal to the principal working direction were obtained for materials not included in the program. For any beltline region materials for which no test material was available, the Charpy USE was conservatively estimated from data obtained on beltline region low-alloy steel forgings, plates, and weld metals.

The fracture toughness properties of bolting materials are discussed in section 3. The requirements and acceptance criteria of Appendix G to 10 CFR 50 are compared to those of ASME Section III. Since the bolting materials ordered before August 16, 1973, meet only the requirements of the applicable ASME Code, it is demonstrated that these materials have adequate toughness for protection against nonductile failure.

The fracture toughness of type 403 modified steel is covered in section 3.3. The test results demonstrate that the material has adequate fracture toughness for protection against failure at 40F. However, B&W specifies a minimum service temperature of 100F for the CRDM, which provides appropriate conservatism.

Section 3.4 describes the supplemental material properties generated to assess the reactor vessel for resistance to ductile tearing instability. These properties are the stress-strain characteristics and the materials resistance to ductile tearing as a function of crack growth. This section also discusses the method of merging the LEFM criteria of section 4 with the EPFM criteria of section 6.

Section 4 describes the basis for the methods used by B&W to obtain the pressure-temperature limit curves. The calculational procedures are primarily based on ASME Appendix G and WRC Bulletin 175. The method of determining the pressure-temperature limit is described for each loading condition of interest.

In section 5, the methods presented and described in sections 3 and 4 are applied to a typical 177 FA plant. Figures in section 5 illustrate (1) the development of the composite limit curves for each loading condition of interest, (2) the development of the reactor criticality limit curve, (3) the limit curves appearing in the Technical Specifications for a typical plant, and (4) the pressure-temperature limits for the preservice system hydrostatic test.

In section 6 the methods are described for qualifying the reactor vessel in the event that a Charpy upper shelf energy of 50 ft-lbs is not obtained. This section determines the pressure limits for ductile tearing instability and states the acceptance criteria. The basis of this analysis is the J-integral and the supplemental fracture toughness data described in paragraph 3.4.

As described in this report, the fracture toughness requirements imposed on the ferritic materials of pressure-retaining components of the RCPB of B&W reactor coolant systems are in compliance with the fracture toughness requirements of Appendix G to 10 CFR 50. In addition, the report demonstrates that the ferritic materials ordered before the effective date of Appendix G to 10 CFR 50 have adequate toughness for protecting against nonductile failure when the system is operated in compliance with the pressure-temperature limit curves developed by B&W. The analytical method employed by B&W to calculate the maximum allowable pressure of the RC system as a function of fluid temperature includes all the margins of safety required by Appendix G.

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December 1984

B&W OWNERS GROUP

Materials Committee

Babcock & Wilcox Owner's Group 177-Fuel Assembly
Reactor Vessel and Surveillance Program
Materials Information

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December 1984

BABCOCK & WILCOX OWNERS' GROUP 177-FUEL ASSEMBLY REACTOR VESSEL AND SURVEILLANCE PROGRAM MATERIALS INFORMATION

by

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SUMMARY

This report is a source document for materials information pertaining to reactor vessels and surveillance programs of the Babcock & Wilcox (B&W) Owners' Group. Due to radiation damage, the mechanical properties of the reactor vessel beltline regions degrade during plant lifetime. The unirradiated and irradiated materials properties of the beltline materials must be known in order to adjust the pressure-temperature limits by which the vessel is operated. This knowledge is acquired by means of a reactor vessel material surveillance program, which provides test specimens of the beltline materials in both unirradiated and irradiated conditions. In order to use test specimen data to represent the vessel materials, the sources and conditions of the materials involved must be known to ensure similarity and applicability. This report provides the information needed to ensure this for the B&W Owners' Group Integrated Reactor Vessel Materials Surveillance Program so far as the information is available.

Materials information for both reactor vessel beltlines and surveillance specimens is detailed in this report. The components and materials have been identified, located, and verified. The chemical compositions of all materials have been reviewed; a convention regarding which analysis to use when more than one is available is established. The material heat treatments are listed, and possible differences between vessel materials and various test specimen sets are described. The orientation of all surveillance program test specimens was checked, and a consistent orientation convention is established. The mechanical test results are presented, along with a position on which set of data to use where more than one is available.

The information presented in this report is B&W's best information and judgment regarding the location and conditions of reactor vessel and surveillance program test specimen materials.

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1. INTRODUCTION

1.1. General

During the working lifetime of a nuclear reactor vessel, the mechanical properties of the ferritic beltline region materials change due to neutron radiation damage. Periodic adjustments in the vessel's pressure-temperature limits, which are used in licensing the vessel, are necessary to reflect the properties changes. To ensure that the pressure-temperature limits provide for safe, but not unduly restrictive, operation of the reactor vessel, the mechanical properties of the irradiated vessel beltline materials must be known. This is accomplished by placing surveillance capsules containing beltline region material test specimens within the reactor vessel, where they are irradiated at a faster rate than the vessel wall. Periodic withdrawal and testing of these specimens monitor the changes of the vessel's mechanical properties in advance of those changes.

The radiation damage can cause radical changes in the vessel material's mechanical properties. Tension tests show substantial increases in yield and tensile strengths, but reduced ductility properties. Charpy impact tests indicate that upper shelf energy decreases, the onset of the upper shelf region is displaced to higher temperatures, and the transition temperature range (from lower shelf to upper shelf) is broadened. Fracture toughness tests show a reduction in toughness in the fully ductile region, along with a displacement to higher temperatures of the fully ductile region. Ideally, these changes would be monitored by capsules containing sufficient numbers and types of test specimens irradiated within the vessel from which the test specimens were drawn.

The ideal is not always realized, however, since changing regulations and standards, based on increased knowledge and experience, have enhanced the desired quality of surveillance programs. While a surveillance program for a new vessel can be altered to meet the augmented requirements, this is not always possible with a program already in place. Changes which have affected or can affect the Babcock & Wilcox (B&W) Owners' surveillance programs include increases in the minimum number of test specimens for each material, requirements that actual, rather than representative, vessel material be encapsulated, and variations in the definition of limiting vessel material. New technology has made it desirable to have fracture toughness test specimens in all capsules; this, of course, cannot be retrofitted. Finally, operating damage to the original surveillance capsule holder tubes (SCHTs) in six vessels of B&W Owners has made it impractical to irradiate surveillance capsules in the actual vessels that they are to monitor. Problems such as these made necessary the development of the Integrated Reactor Vessel Materials Surveillance Program (IRVSP) concept.

The B&W Owners' Group IRVSP addresses three major issues. To solve the problem of failed holder tubes, the IRVSP created "guest" and "host" reactors, where one vessel's capsules are irradiated in another vessel. Since all the vessels in the IRVSP have the same geometry and operate at similar power levels and flux spectra, there are essentially no operating differences from one vessel to another. This makes it feasible to relocate a capsule from its own vessel to a "host" vessel. To resolve the issue of test specimens not corresponding to the vessel's materials, the data from all the Reactor Vessel Surveillance Programs (RVSPs) will be integrated so that specimens from one vessel's capsules can be applied to another vessel's materials, or a "pool" of similar material data can be created so that a lower bound set of mechanical properties can be developed for all vessels. Several research capsules have been fabricated to expand the data base and take advantage of new technology. These capsules contain weld metal test specimens (including fracture toughness test specimens) drawn from excess surveillance weld metal or nozzle belt dropouts. These capsules are irradiated in the host reactors in the same SCHTs as the surveillance capsules.

The IRVSP concept increases the need for accurate documentation of the vessel and surveillance program materials. In order to determine which test specimen can apply to which vessel, the type, source, chemical composition, orientation, processing history, and heat treatment history of the materials must be known for both the specimen and the vessel. While this information is not usually a problem for a plant-specific RVSP, where such parameters are supposed to be equivalent for both the vessel and test specimens, it can be a concern in an IRVSP, since fabrication processing can vary from one vessel (and RVSP) to another.

1.2. Objective

This report is a source document for the B&W Owners' Group reactor vessels and surveillance programs. The materials in the vessels and surveillance programs have been verified by referring to original records, as-built construction diagrams, mill test reports, and weld qualification documents. The chemical compositions of all materials have been verified by the review of test reports, weld qualification reports, and B&W testing; a convention regarding which composition to use when more than one analysis is available is established. The heat treatments of the various materials are listed, as are the unirradiated mechanical properties. The orientation of all test specimens was verified by referring to the applicable sectioning diagrams, and an orientation convention is established.

Engineering judgment was required to interpret some of the available records, since early (mid-1960s) documentation requirements were not according to current standards. These judgments pertain mainly to material heat treatments and resolving material location and/or identification discrepancies, which arose initially due to typographical or copying errors. The information presented in this report is B&W's best data regarding the locations and conditions of vessel and test specimen materials.

2. BACKGROUND

2.1. Reactor Vessel Construction

A main point of this report is the difference in behavior, due to variations in fabrication, of the vessel materials and multiple sets of test specimens. To understand these differences it is helpful to have a basic knowledge of the fabrication processing steps in reactor vessel construction.

For the purposes of this report, the processing began while the heats of steel were still molten in the steelmaking furnace. For each heat, a sample of molten metal was drawn off, solidified, and analyzed; this gave a ladle analysis of the heat. The heat was then cast into an ingot and assigned a heat number by the steelmaker. The processing may then have followed two courses, depending on whether plate or a forging was produced, since the core region shell courses were of either plate of forging construction.

If plate was produced, the ingot was heated and hot rolled to a final thickness of approximately 8-3/4 inches. At some point during the rolling process, the ingot was separated into two or more pieces; this accounts for sections of plate having heat numbers such as C4344-1. The plate was subsequently heat treated, usually by the normalizing, austenitizing, quenching, and tempering stages. Excess material was cut from the plate and stress relieved. From this material, test specimens were machined, and their test results were reported on the material supplier's mill test reports. At some time after the rolling process began, a second chemical analysis, this time on a sample cut from the solid product, may have been conducted; this is called a product analysis. The plate was then shipped to B&W's Mt. Vernon works.

After receipt at Mt. Vernon, the plate was rolled into a semicylindrical shape and then electroslag welded to another rolled plate to form a shell. The shell was loaded into a furnace and heat treated again, using processing stages similar to those of the steelmaker. After heat treatment, the electroslag welds were cut out of the shell, and excess (offal) material was cut off the top or bottom edge. Some of this excess material was rough-machined, stress-relieved, then final machined into test specimens which provided data for Mt. Vernon's qualification tests. Other excess material may have been rough-machined later, given a different stress relief (weld metal and heat-affected zone specimens were submerged-metal arc welded prior to stress relief), and final machined into test specimens for the RVSP of that plant. The vessel plate was prepared for welding, then automatic submerged-metal arc (ASA) welded to the other plate to form a shell once again. The shell received a stress relief treatment, then was ASA welded into place in the vessel. At least one more stress relief followed this last welding stage.

If a forging was produced, the ingot was hot-ring forged to a thickness of approximately 8-3/4 inches. The steelmaker usually assigned a forging number in addition to the heat number previously assigned. Excess material was cut from the forging, and both the forging and excess material were heat treated, usually with the normalizing, austenitizing, quenching, and tempering stages. Some excess material was rough-machined and stress-relieved. From this material mechanical test specimens were machined, and their test results were included in the supplier's mill test reports. At some point a product analysis was run. The forging was final machined and, along with the excess material, was shipped to Mt. Vernon.

After receipt at Mt. Vernon, the forging was ASA welded into place in the vessel. At least one stress relief followed the welding stage. Some of the excess material may have been rough-machined, stress-relieved, then machined into test specimens which provided data for Mt. Vernon qualification tests. Other excess material later may have been rough-machined, given a different stress relief (weld metal and heat-affected zone specimens were ASA welded prior to stress relief), and machined into test specimens for the RVSP of that plant.

ASA welds in both the reactor vessels and surveillance programs are designated by numbers such as SA 1585 or WF 232. These numbers were assigned to specific combinations of weld filler wire heats and weld flux lots at the time a weld qualification test of that combination was conducted at either the Mt. Vernon or Barberton works. All ASA welds that were made with the same filler wire heat/flux lot combination as a weld qualification test were identified by that weld qualification number. While two different welds with the same number therefore have the same source material, differences in welding parameters and heat treatments mean that similar mechanical behavior is not assured. These variables should be considered in any data analysis, although their effect on material properties is usually minor.

2.2. Reactor Vessel Sections of Concern

Any material in a reactor vessel that receives significant neutron radiation exposure should be evaluated for degradation of materials properties since it may be the limiting or "worst-case" component of the vessel. The most heavily neutron irradiated section of a reactor vessel is the beltline region, which encompasses all materials that must be evaluated.

The current (effective May 27, 1983) revision to 10 CFR 50, Appendix G[1] defines the beltine region as

> . . . the region of the reactor vessel (shell material including welds, heat-affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

This replaced an earlier definition which limited the "adjacent regions" to those which were expected to experience a shift in reference temperature (ΔRT_{NDT}) of 50F or greater during plant lifetime. For practical purposes, the materials which need to be considered are 1) the upper shell, the lower shell and, in the case of Oconee 1, the intermediate shell; 2) any longitudinal welds in these shells; and 3) the circumferential welds connecting these shells. To allow for possible future changes in the beltline definition and for unforeseen circumstances, the lower nozzle belt forging (except for Davis-Besse 1, which has its only nozzle belt forging in the beltline, the dutchman forging, the nozzle belt-to-upper shell circumferential weld (except for Oconee 1, where it is the nozzle belt-to-intermediate shell circumferential weld), and the lower shell-to-dutchman forging circumferential weld are included as beltline materials in this report.

2.3. Material Chemical Composition Clarification

The chemical compositions of the reactor vessel and surveillance program materials, aside from verifying the type of material in use, affect how those materials perform in service and are also inputs into various predictive models used to determine radiation-induced mechanical properties degradation. Some of these models are used to estimate material properties upon which licensing restrictions are based. There are different sources of chemical analysis for base metal (plate for forgings) and weld metal.

For plate or forgings, there are two basic types of analysis. The first is a ladle, or heat, analysis, which is run on a sample of metal drawn from a molten heat of steel. The other is a product, or check, analysis, which is run on a sample cut from the solidified steel product. It is B&W's position that a product analysis is more reliable than a ladle analysis and should be stated as the material composition whenever one is available. If more than one analysis of a particular type exists, an average should be used.

For weld metal, there are four possible sources of analysis. These are drawn from (listed in descending order of reliability) BAW-1500, Table 5; BAW-1500, Table 10; BAW-1500, Table 6; and the weld qualification analysis. B&W report BAW-1500[2] covers an extensive re-analysis of available archive weld metals drawn from surveillance programs and nozzle belt dropouts; many analyses were performed on each weldment. When the weld qualifications were run, one analysis was run on a test weldment. Table 5 of BAW-1500 lists the average chemical composition for each specific weldment analyzed. A weld such as WF 209 is listed more than once, since one WF 209 weld was made for the Oconee 2 surveillance program, another WF 209 weld was made for Oconee 3, etc; each time WF 209 is indicated, a different analysis accompanies it. Only surveillance block welds can have a Table 5 analysis, since no actual beltline welds have been analyzed.

Table 10 of BAW-1500 lists a "best judgment" composition for reactor vessel welds. These compositions are based on actual analyses of welds made of the same weld wire heat (and in some cases the same wire/flux combination) as the weld identified with the composition. There is some engineering judgment associated with this table, since the designated chemical compositions were not obtained from the actual welds.

Table 6 of BAW-1500 lists average chemical compositions grouped by weld wire heat. All welds made with the same wire heat, regardless of flux lot, are grouped together. Since the composition of a weld deposit is determined primarily by the wire composition, this table provides useful approximations when Table 5 or Table 10 analyses are not available.

The weld qualification report chemical compositions were determined by the analysis of test weldments made when weld wire lots were initially received at the B&W manufacturing facility. Generally, only one analysis was run on each weldment. Since later testing is considered more reliable, weld qualification analyses should only be used if better analyses are not available.

To summarize, a product analysis should be considered more accurate than a ladle analysis for plate or forgings. For welds, BAW-1500 Table 5 is considered more accurate than Table 10, Table 10 is considered more accurate than Table 6, and Table 6 is considered more accurate than the weld qualification analysis. For licensing purposes, the most accurate analysis available for a material is used.

2.4. Test Specimen Orientation

In this section, the orientation of test specimens with respect to their source sections is reviewed and an orientation convention defined. This convention is necessary to eliminate confusion regarding the currently used terminology -- confusion which is due partially to the lack of a single standard covering the situation.

Two American Society for Testing and Materials (ASTM) standards address the issue of specimen orientation. Standard A 370[3] establishes one convention for tension, bend, and impact tests, based on rolling direction (or extension), and applicable to general wrought products (see Figure 2.4-1). The orientations given are longitudinal and transverse, and describe the direction of the axis of the test specimen with respect to the rolling direction. No provision is made for a variance in the direction of the notch of the impact specimen. Figure 2.4-2, also from A 370, shows test specimen locations taken from forgings, with orientations given as longitudinal, radial, and tangential. These orientations are based on the axis of the forging and make no mention of principal working direction. It is common practice to use the terms "transverse" and "tangential" interchangably; however, this is only correct if the axis of the forging is also the principal working direction. B&W shell course forgings are expanded outward, making the principal working direction around the circumference, i.e., in the tangential direction. In this case, a tangential test specimen from a forging corresponds to a plate longitudinal specimen. To eliminate the potential for confusion, the forging orientations shown in Figure 2.4-2 will not be used to describe B&W test specimens.

Standard E 399[4] establishes a convention for fracture toughness specimens with a crack, as shown in Figure 2.4-3. This standard accounts for the orientation of the crack, but also identifies the forging axis with the rolling direction. It does not specifically mention Charpy impact specimens, although it has become conventional to use "TL" and "LT" to describe Charpy directions.

The convention to be described will incorporate aspects of both A 370 and E 399. The principal working direction of the material, whether plate or forging, will be the key to test specimen orientation. The "longitudinal" and "transverse" descriptions from A 370 will be retained, as will the "TL," "LT," and "TS" type descriptions of crack orientation from E 399, which will be expanded to include impact (Charpy) specimens. The results will be a consistent system which will apply to all B&W Owners' RVSP specimens.

Surveillance test specimens are taken from excess shell course material, as shown in Figure 2.4-4. The principal working direction is circumferential (around the barrel), regardless of whether plate or forging is used. The S, T, and L directions are adopted here to be consistent with E 399; this is not consistent with previous B&W drawings.[5] The excess shell material is cut into sections of manageable size, as shown in Figure 2.4-5. From this step, two courses of action are possible. The first, for base material, is shown in Figure 2.4-6; the block is sectioned into two pieces, each approximately 2-1/2 x 6 x 8 to 15 inches, corresponding to the 1/4 and 3/4 thickness (T) locations in the plate. A later revision of this process differs slightly; a single 5-inch deep section is cut from the center of the block, and is then sectioned into two 2-1/2-inch deep pieces. The second course, for weld and heat-affected zone (HAZ) material, is shown in Figure 2.4-7. In this case, the inside and outside diameter surfaces of two blocks are machined away, leaving rectangular sections. These two sections are then prepared in a V-groove arrangement and welded using the ASA process; the procedures used are the same as those for the reactor vessel welds. The weld produced is equivalent to a circumferential weld in a vessel. The 1/4 T and 3/4 T location pieces are then cut as shown.

The final sectioning for specimens is described in the Lynchburg Research Center (LRC) surveillance capsule construction reports.[6-13] When the information presented in this section is combined with the LRC sectioning diagrams, it is possible to determine specimen orientation.

Figure 2.4-8a shows a typical sectioning diagram for base metal. The orientation of the plate is shown in Figure 2.4-8b. Four different test specimens were machined from base material for the RVSPs; these are the transverse and longitudinal tension specimens, and the TL and LT Charpy specimens.

Figure 2.4-9a shows a typical sectioning diagram for HAZ material. The orientation of the material is shown in Figure 2.4-9b. Four different test specimens were machined from HAZ material for the RVSPs: the longitudinal tension specimen, which is parallel both to the principal working direction and the axis of the weld, and which is entirely in the base metal portion of the block; the transverse tension specimen, which is perpendicular both to the principal working direction and the axis of the weld, and which is partly in weld metal and partly in base metal; the TL (transverse) Charpy specimen, which is perpendicular both to the principal working direction and the axis of the weld, is notched in the TL orientation, and which is partly in weld metal and partly in base metal; and the TS (transverse) Charpy specimen, which is perpendicular both to the principal working direction and the axis of the weld, is notched in the TS orientation, and which is partly in weld metal and partly in base metal. The TS (transverse) Charpy specimens have been called longitudinal in the past; this usage should be discontinued, as it implies a specimen with its axis parallel to the weld axis. The Charpy specimens are identified with respect to both base metal principal working direction (TL or TS) and weld axis (transverse).

Figure 2.4-10a shows a typical sectioning diagram for weld metal. The orientation of the plate is shown in Figure 2.4-10b. Three different test specimens were machined from weld metal for the RVSPs; these are the longitudinal tension specimen, the TL (transverse) Charpy specimen, and the TL (transverse) compact fracture toughness (CT) specimen of various sizes. The "transverse" may be dropped in this case since "TL" describes the orientation sufficiently in all cases.

A summary of the orientation convention is given in Table 2.4-1. Figures 2.4-8b, 2.4-9b, and 2.4-10b should be considered the definitive description of orientations for B&W materials.

Table 2.4-1. B&W Test Specimen Orientations

Specimen type	Material		
	Base metal	HAZ metal	Weld metal
Tension	Longitudinal Transverse	Longitudinal Transverse	Longitudinal
Charpy	LT TL	TL (Transverse) TS (Transverse)	TL
Compact fracture toughness	NA[a]	NA[a]	TL

[a]NA denotes not applicable.

Figure 2.4-1. The Relation of Test Coupons and Test Specimens to Rolling Direction or Extension (Applicable to General Wrought Products) (From ASTM Standard A 370)

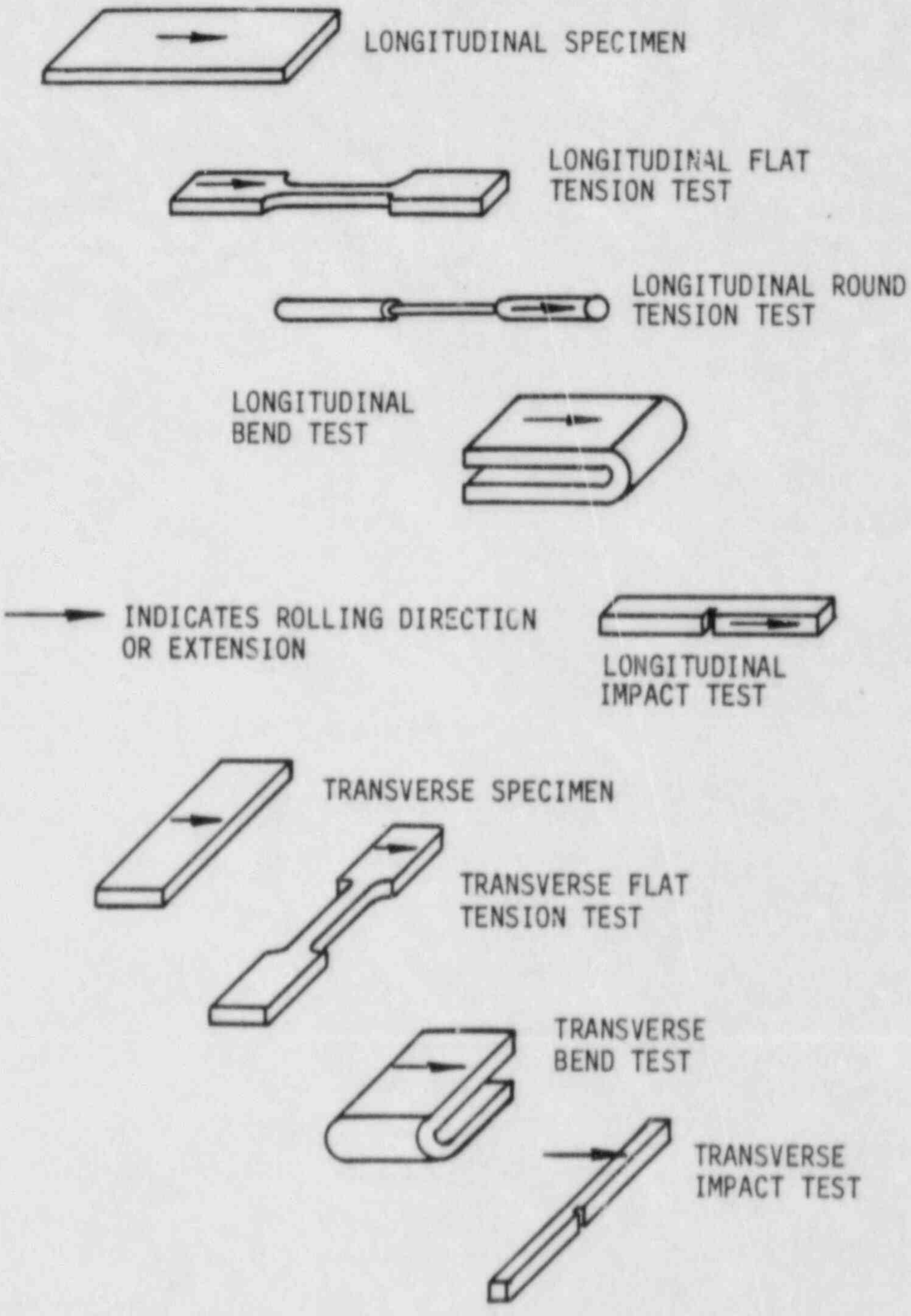
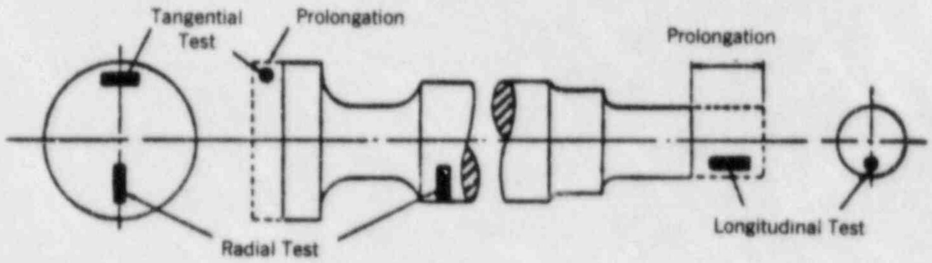
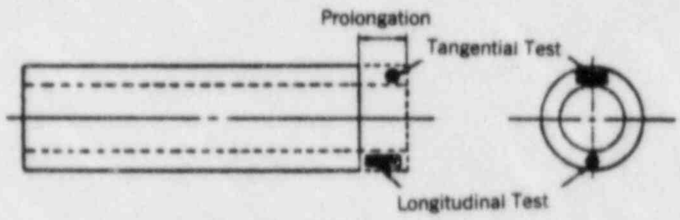


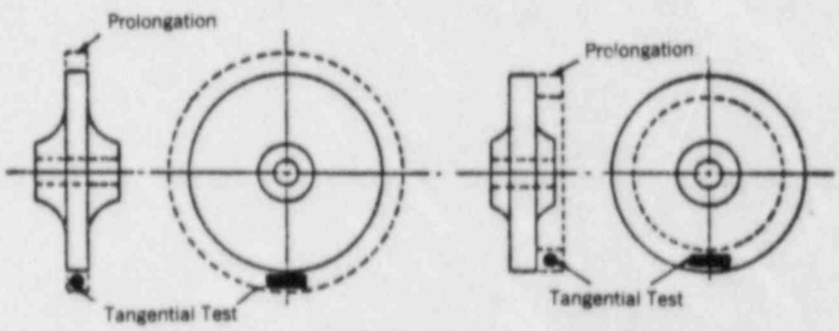
Figure 2.4-2. Location of Test Specimens for Various Types of Forgings (From ASTM Standard A 370)



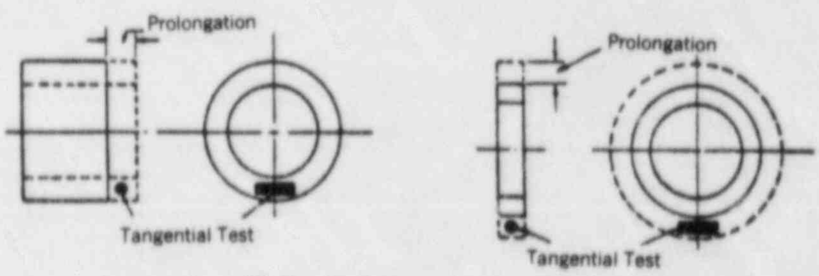
(a) Shafts and Rotors



(b) Hollow Forgings



(c) Disk Forgings



(d) Ring Forgings

Figure 2.4-3. Crack Plane Orientation Code for Rectangular Specimens (From ASTM Standard E 399)

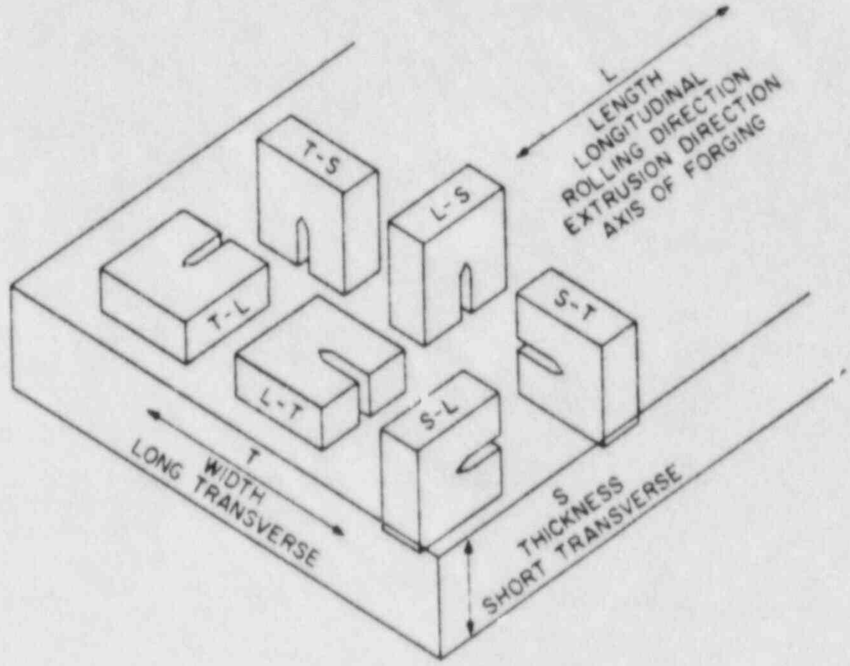


Figure 2.4-4. Typical Shell Course and Excess Material Section

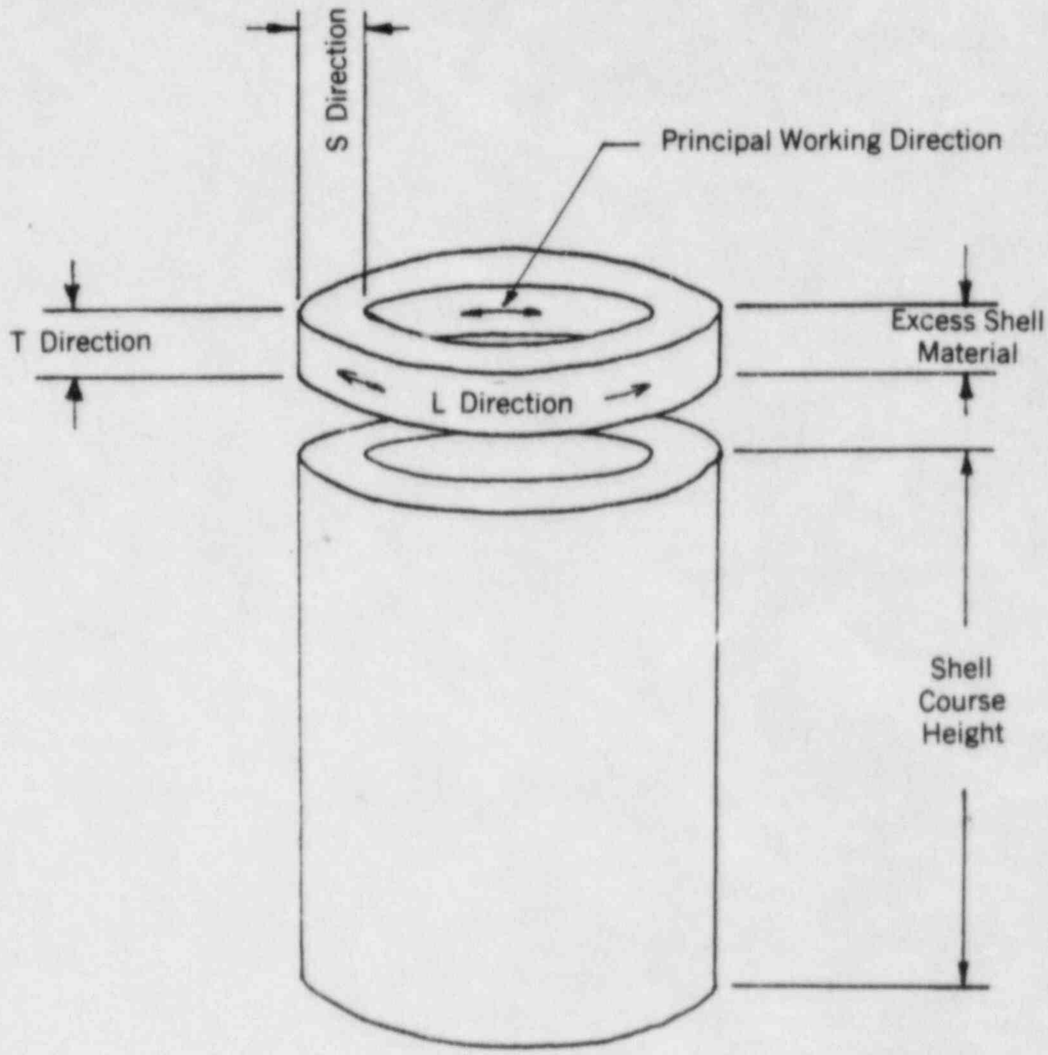


Figure 2.4-5. Typical Excess Material Section -- Rough Cut

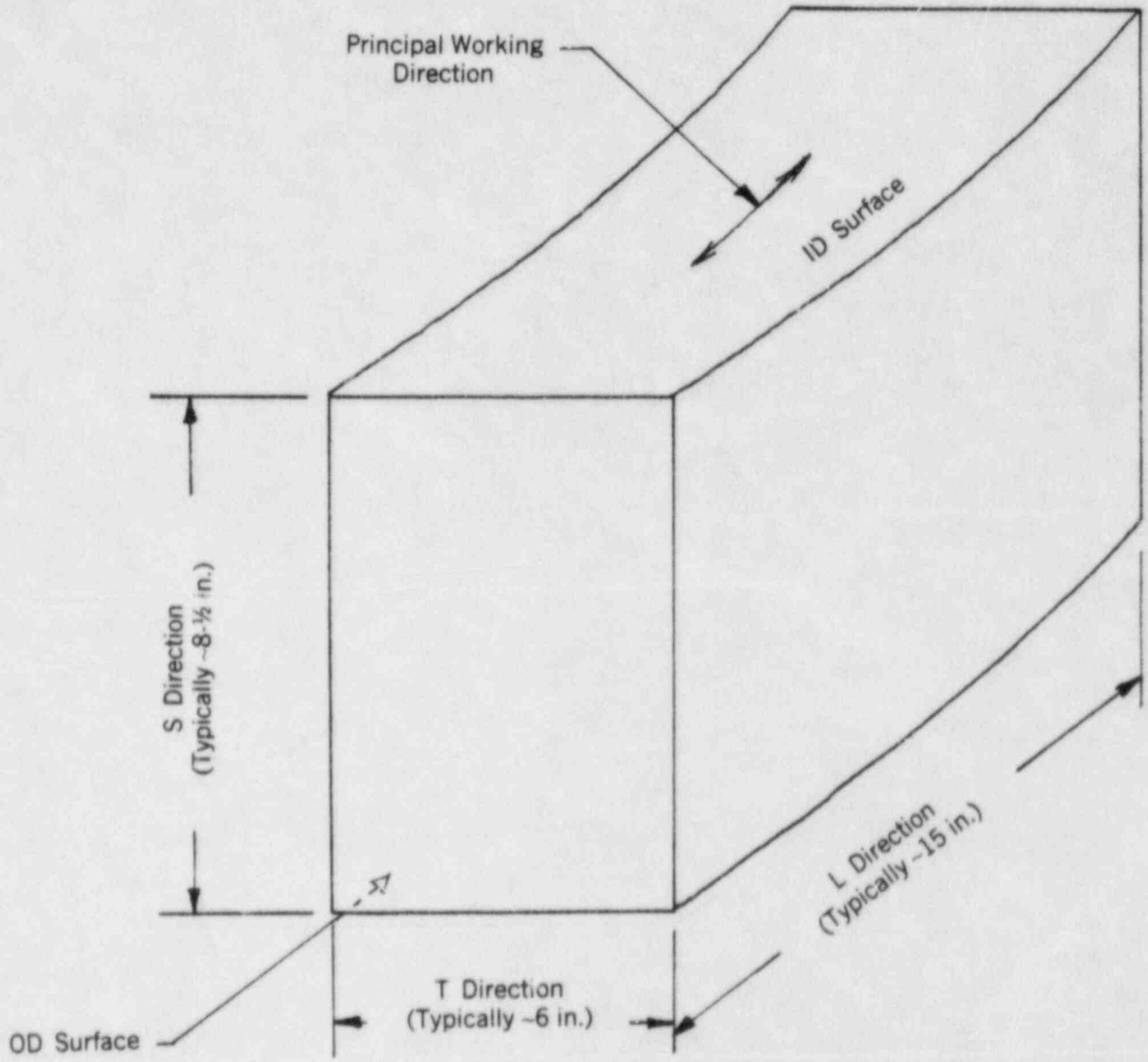


Figure 2.4-6. Base Metal Section for RVSP Test Specimens

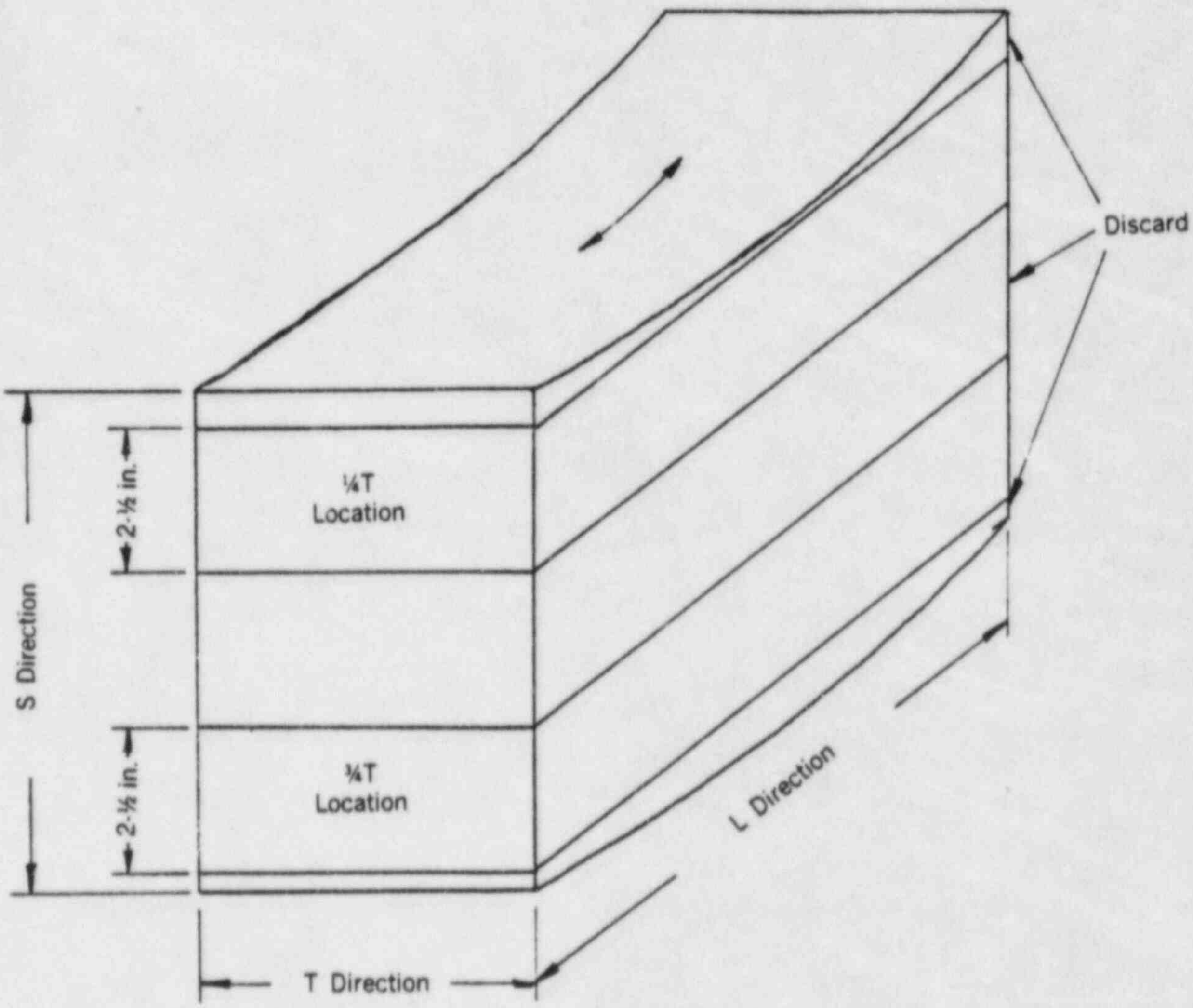


Figure 2.4-7. HAZ or Weld Metal Section for RVSP Test Specimens

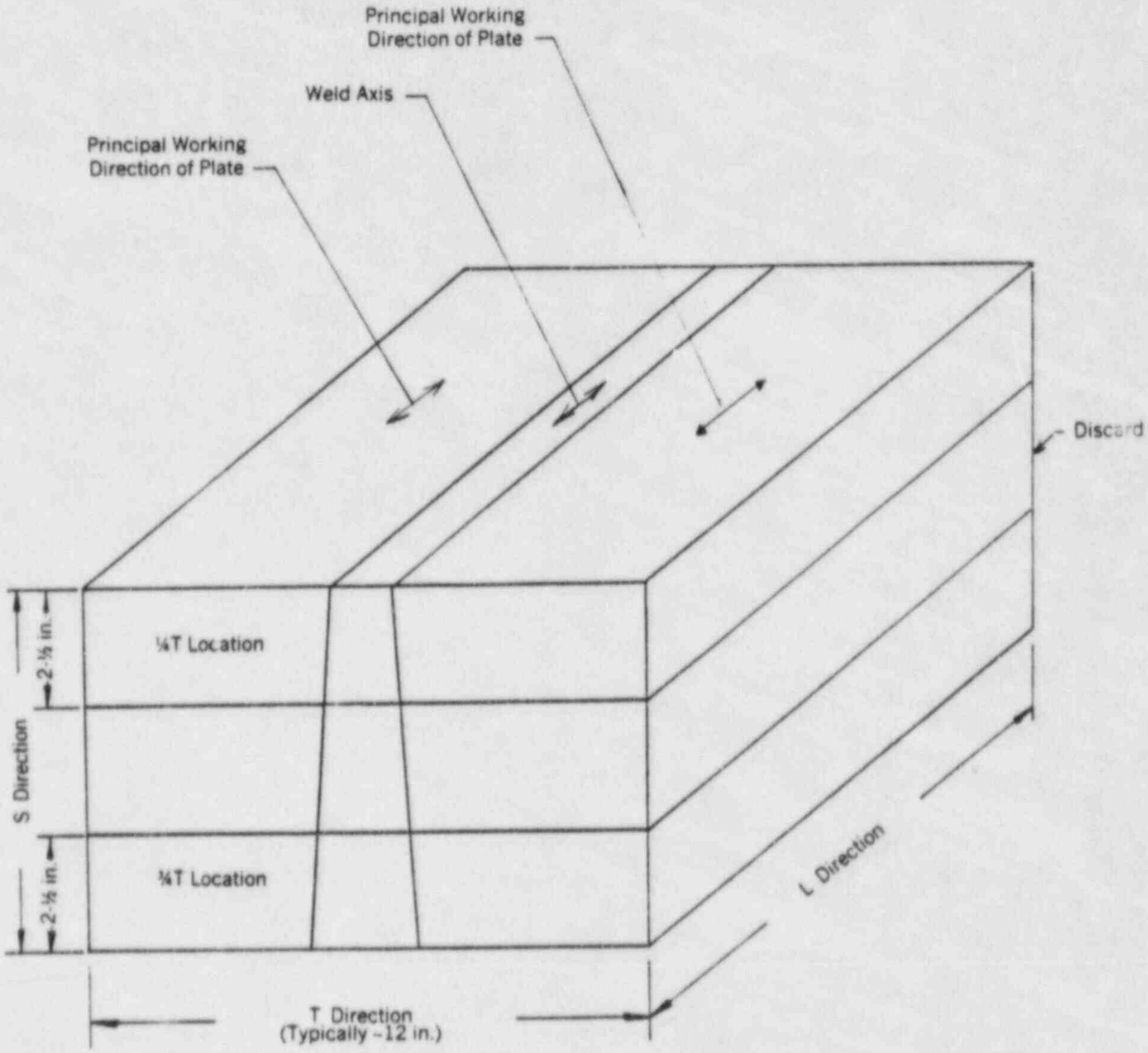
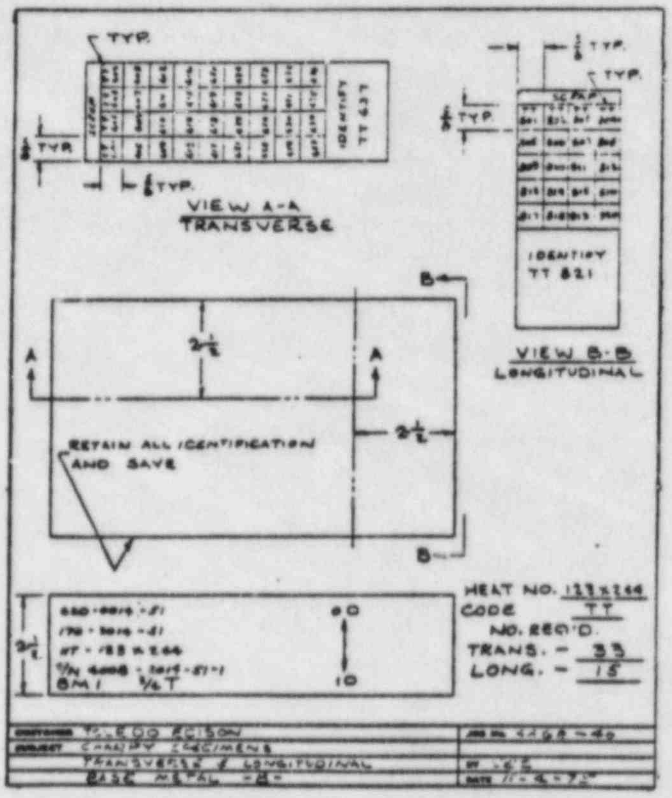
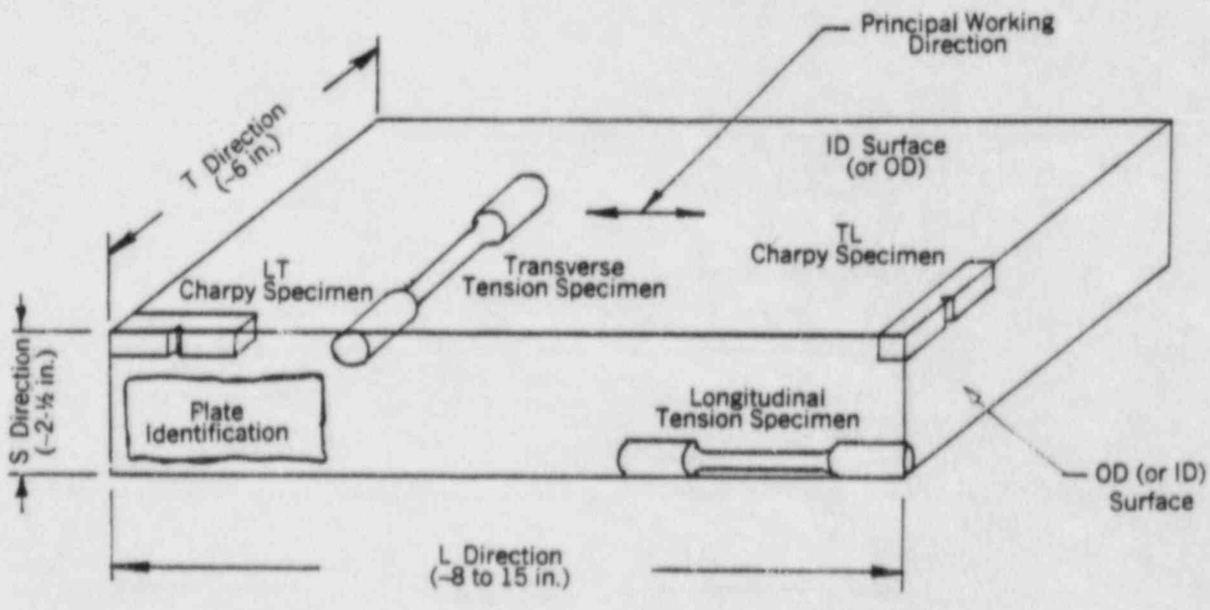


Figure 2.4-8. Base Metal Specimens

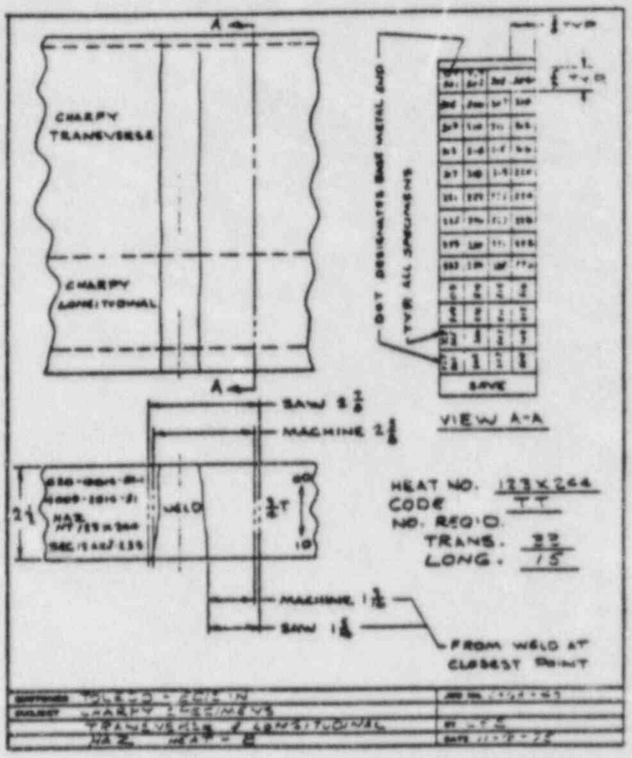


a) LRC Sectioning Diagram

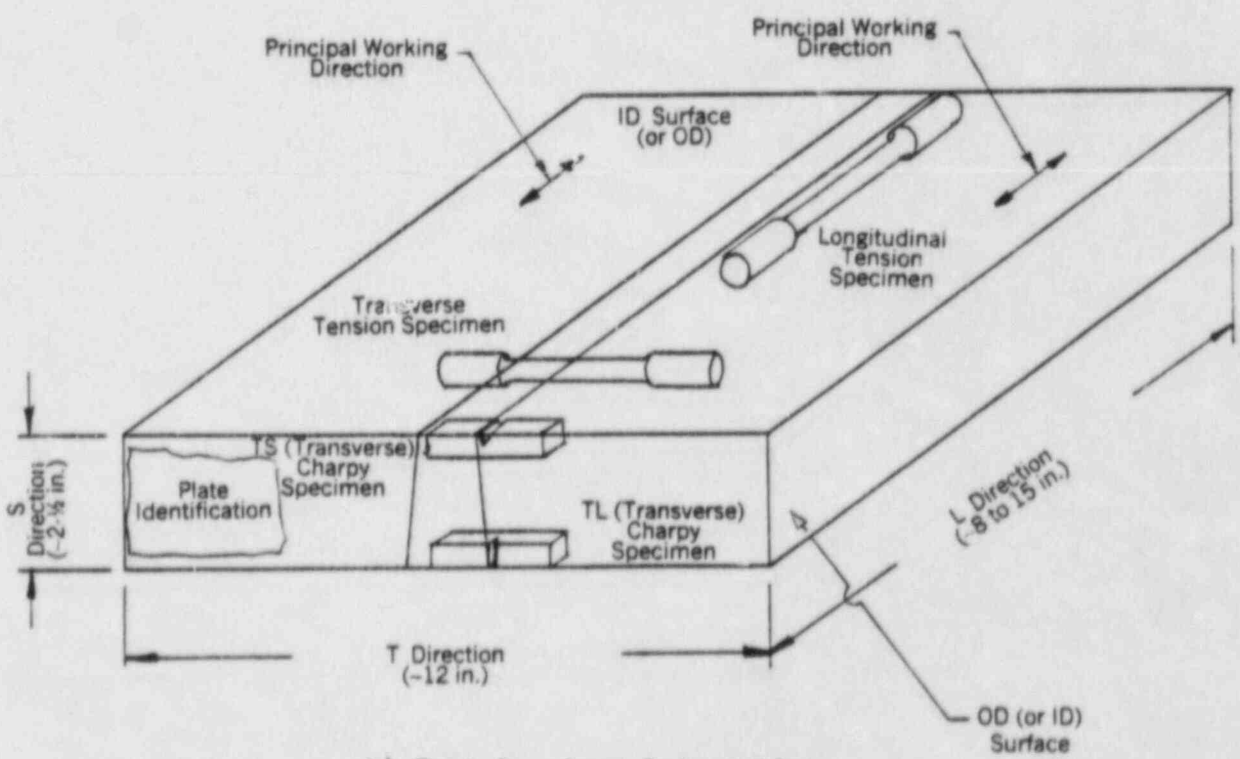


b) Te:t Specimen Orientations

Figure 2.4-9. HAZ Material Specimens

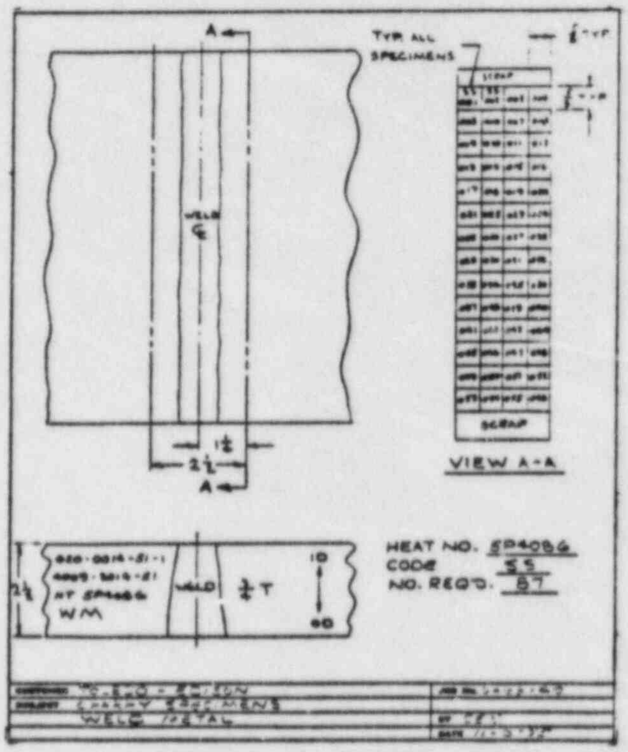


a) LRC Sectioning Diagram

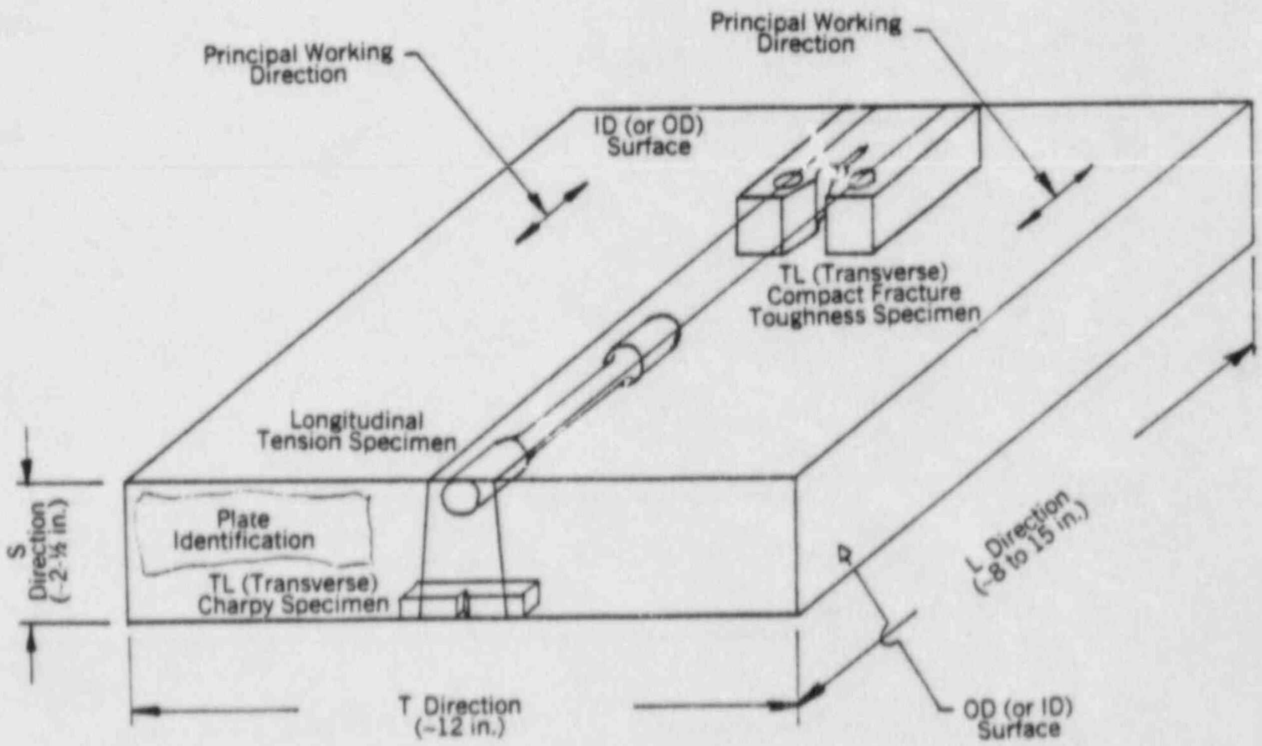


b) Test Specimen Orientations

Figure 2.4-10. Weld Metal Specimens



a) LRC Sectioning Diagram



b) Test Specimen Orientations

3. REACTOR VESSEL INFORMATION

3.1. Description

This section contains information pertaining to the locations of components and welds, chemical compositions, heat treatments, and unirradiated mechanical properties. Each subsection is devoted to the data pertaining to one reactor vessel, with information presented wholly in tabular or graphical format. Tables 3.n.1. present background information on the nuclear plants and on vessel fabrication. Tables 3.n.2. list the chemical compositions for each vessel beltline materials; the choice of licensing analysis should follow the convention developed in section 2.3.

Tables 3.n.3. list the heat treatments for the vessel (not test specimen) materials, so far as they could be determined from the available processing information. Since vessel materials often received several heat treatments at different locations and at different processing stages, it is difficult to reconstruct the paper trail years after fabrication. In the event that this becomes an area of concern, Mt. Vernon fabrication records can be consulted in an effort to clarify the situation. The information presented here is obtained from mill test reports and Mt. Vernon test reports, and is believed to be accurate.

Tables 3.n.4. list the unirradiated mechanical properties for the vessel materials, based on tests run at the time of vessel construction. Two sources, mill test reports and Mt. Vernon tests, may be available for base metal properties. When both sources are available, the Mt. Vernon tests should be considered more reliable than supplier data, since these tests were more extensive, better documented, and test specimens are more likely to match the heat treatments of the vessel materials. In both cases, documentation rarely includes mention of test specimen orientation. Since industry practice during that period called for longitudinal test specimens, all specimens should be considered longitudinal (tension test) or LT (Charpy test) unless otherwise noted.

Surveillance program test data (see section 4) may also be used to describe the properties of the reactor vessel material; these data are better documented than Mt. Vernon test data, and test specimen orientation is known. When both Mt. Vernon test data and surveillance program test data are available for a vessel material, the more conservative data set should be used for licensing purposes.

Usually only weld qualification test data are known for weld metal in the reactor vessels. This information is not extensive, so in the event that Mt. Vernon test data are available, they should be considered more reliable than the weld qualification. When surveillance program test data are available, they should be used for licensing purposes.

In any case where more than one data set is available for the same material (three test areas, for instance), the most conservative data should be used for licensing purposes. When multiple tension tests or chemical analyses are available, the mean of the data should be used for licensing.

Figures 3.n.1, 3.n.2, and 3.n.3 show the locations of the various sections of the reactor vessels.

3.2. Oconee Unit 1 (NSS-3)

Table 3.2-1. General Plant Information for Oconee Unit 1 (NSS-3)

Plant Designation: Oconee Nuclear Station, Unit 1

Plant Location: Seneca, South Carolina

Start-up Date (on power grid): 5/4/73

Owner Utility: Duke Power Company

B&W Contract Number: 620-0003

Reactor Vessel Information

ASME Code Date: 1965 edition including Addenda through Summer 1967

Fabrication Site: Barberton, Ohio and Mt. Vernon, Indiana

Fabrication Schedule:

Start Date -- 9/21/67
Ship Date -- 12/15/69

Vessel Materials:

Nozzle Belt Forgings -- A 508 Cl.2 modified by ASME Code Case 1332-2
Beltline Base Materials -- SA-302B modified by ASME Code Case 1339
Beltline Weld Materials -- ASA weld; MnMoNi wire, Linde 80 flux

Location of Materials in Vessel: See Figure 3.2-1.

Chemical Composition of Materials: See Table 3.2-2.

Heat Treatment of Materials: See Table 3.2-3.

Vessel Stress Relief: Upper Shell -- 1100-1150F for 50h (cumulative)
Lower Shell -- 1100-1150F for 49h (cumulative)

Table 3.2-2. Chemical Composition of Beltline Region Materials for Oconee Unit 1

Material ID	Chemical Composition, wt %: C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
AHR 54; ZV 2861	0.18*	0.64	0.006	0.010	0.29	0.65	0.31	0.57	0.155	0.01 V, 0.01 Co Check analysis
	0.21	0.63	0.008	0.007	0.27	0.70	0.38	0.58		0.01 V, 0.01 Co Ladle analysis
C2197-2	0.21*	1.28	0.008	0.010	0.17	0.50		0.46		0.021 Co Ladle analysis
	*								0.15	Check analysis
C3265-1	0.21	1.42	0.015	0.015	0.23	0.50	0.17	0.49	0.10	0.016 Co
C3278-1	0.19	1.26	0.010	0.016	0.23	0.60	0.11	0.47	0.12	0.016 Co
C2800-1	0.20	1.40	0.012	0.017	0.20	0.63	0.13	0.50	0.11	0.014 Co
C2800-2	0.20	1.40	0.012	0.017	0.20	0.63	0.13	0.50	0.11	0.014 Co
122S347VA1	0.20#	0.63	0.010	0.008	0.25	0.66	0.32	0.55		0.02 V, 0.021 Co Check analysis
	0.20#	0.62	0.010	0.009	0.25	0.67	0.33	0.57		0.02 V, 0.025 Co Check analysis
	0.21	0.60	0.010	0.008	0.24	0.70	0.33	0.60	0.08	0.01 V, 0.010 Co Ladle analysis
SA 1135	0.08*	1.45	0.011	0.013	0.49	0.54	0.08	0.38	0.25	BAW-1500, Table 10
	0.08	1.44	0.012	0.011	0.51	0.54	0.08	0.38	0.25	BAW-1500, Table 6
	0.08	1.24	0.015	0.013	0.47	0.50	0.06	0.54	0.17	Weld qualification

Table 3.2-2. (Cont'd)

Material ID	Chemical Composition, wt %								Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
SA 1229	0.06*	1.56	0.021	0.012	0.43	0.61	0.16	0.37	0.26	BAW-1500, Table 10
	0.08	1.51	0.020	0.013	0.57	0.61	0.16	0.37	0.25	BAW-1500, Table 6
	0.06	1.56	0.021	0.012	0.43	0.57	0.17	0.43	0.20	Weld qualification
WF 25	0.09*	1.60	0.015	0.016	0.50	0.68	0.09	0.42	0.35	BAW-1500, Table 10
	0.09	1.58	0.014	0.016	0.53	0.68	0.09	0.42	0.35	BAW-1500, Table 6
	0.088	1.50	0.019	0.010	0.45	0.71	0.11	0.33	0.29	Weld qualification
SA 1585	0.08*	1.45	0.016	0.016	0.51	0.59	0.09	0.38	0.21	BAW-1500, Table 10
	0.08	1.45	0.016	0.016	0.51	0.59	0.09	0.38	0.21	BAW-1500, Table 6
	0.08	1.35	0.016	0.011	0.43	0.51	0.06	0.35	0.25	Weld qualification
WF 9	0.08*	1.45	0.016	0.016	0.51	0.59	0.09	0.38	0.21	BAW-1500, Table 6
	0.076	1.30	0.015	0.012	0.55	0.60		0.55	0.174	Weld qualification
SA 1073	0.10*	1.38	0.025	0.017	0.51	0.64	0.11	0.43	0.29	BAW-1500, Table 10
	0.10	1.38	0.025	0.017	0.51	0.64	0.11	0.43	0.21	Weld qualification
SA 1493	0.08*	1.51	0.017	0.010	0.46	0.55	0.12	0.41	0.29	BAW-1500, Table 10
	0.08	1.51	0.017	0.010	0.46	0.43	0.12	0.45	0.22	Weld qualification
SA 1430	0.08*	1.43	0.017	0.015	0.43	0.55	0.12	0.41	0.29	BAW-1500, Table 10
	0.08	1.43	0.017	0.015	0.43	0.60	0.12	0.42	0.16	Weld qualification
SA 1426	0.08*	1.53	0.017	0.013	0.43	0.55	0.12	0.41	0.29	BAW-1500, Table 10
	0.08	1.53	0.017	0.013	0.43	0.61	0.12	0.36	0.18	Weld qualification

*Analysis to be used for licensing.
#Average of multiple analysis to be used for licensing.

Table 3.2-3. Heat Treatment of Beltline Region Materials for Oconee Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
AHR54; Lower nozzle belt forging	ZV 2861	NA	A 508-64 Cl.2, mod. to ASME Code Case 1332-2	Ladish	1650F ±20F for 8h, FC; 1200F ±20F for 10h, AC; 1600F ±20F for 5h, WQ; 1250F ±20F for 15h, WQ
C2197-2 Intermediate shell	C2197-2	NA	SA-302B, mod. to ASME Code Case 1339	Lukens	1650-1700F for 1h/in. WQ; 1200-1250F for 1h/in. AC. Probably also 1675-1725F for 6-1/2h, BQ; 1600-1650F for 9-1/2h, BQ; 1200-1225F for 9-1/2h, BQ.
C3265-1 Upper shell	C3265-1	NA	SA-302B, mod. to ASME Code Case 1339	Lukens	1625-1675F for 1h/in. WQ; 1200-1250F for 1h/in. AC. Probably also 1600-1650F for 9-3/4h, BQ; 1200-1220F for 9-1/2h, BQ.
C3278-1 Upper shell	C3278-1	NA	SA-302B, mod to ASME Code Case 1339	Lukens	1625-1675F for 1h/in. WQ; 1200-1250F for 1h/in. AC. Probably also 1600-1650F for 9-3/4h, BQ; 1200-1225F for 9-1/2h, BQ.
C2800-1 Lower shell	C2800-1	NA	SA-302B, mod. to ASME Code Case 1339	Lukens	1625-1675F for 1h/in. WQ; 1200-1250F for 1h/in. AC. Probably also 1600-1650F for 9-1/2h, BQ; 1200-1225F for 9-1/2h, BQ.
C2800-2 Lower shell	C2800-2	NA	SA-302B, mod. to ASME Code Case 1339	Lukens	1625-1675F for 1h/in. WQ; 1200-1250F for 1h/in, AC. Probably also 1600-1650F for 9-1/2h, BQ; 1200-1225F for 9-1/2h. BQ.

Table 3.2-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
122S347VA1 Dutchman	122S347VA1	NA	A 508-64 Cl.2, mod. to ASME Code Case 1332-2	Bethlehem	1550F for 15h, WQ; 1200 for 16h, AC.
SA 1135 NB to IS Circ. weld	61782	Linde 80 8457		Amer. Chain & Cable	
SA 1229 IS to US Circ. weld	71249	Linde 80 8492		Amer. Chain & Cable	
WF 25 IS to US Circ. weld	299L44	Linde 80 8650		Union Carbide	
SA 1585 US to LS Circ. weld	72445	Linde 80 8597		Amer. Chain & Cable	
WF 9 LS to Dutch Circ. weld	72445	Linde 80 8632		Amer. Chain & Cable	
SA 1073 IS Long. weld	1P0962	Linde 80 8445		US Steel	
SA 1493 US Long. weld	8T1762	Linde 80 8578		US Steel	

Table 3.2-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
SA 1430 LS long. weld	8T1762	Linde 80 8553		US Steel	
SA 1426 LS long. weld	8T1762	Linde 80 8553		US Steel	

Note: Welds and plates may have received intermediate stress reliefs. All received full vessel stress relief. Upper shell cumulative received 50h at 1100-1150F. Lower shell cumulative received 49h at 1100-1150F. US to LS circ. weld stress relief believed to be either 31 or 40 h, based on interpretation of RVSP weld records.

[a] NA denotes not applicable.
[b] FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Table 3.2-4. Unirradiated Mechanical Properties of Beltline Region Materials for Oconee Unit 1

Material ID	Heat treatment	Toughness Properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile Properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
AHR54	As per 3.2-3, plus		87							61.5	82.3	31.0	75.0	Supplier Test Report data
	1125F ±25F for		54							67.6	89.0	30.0	73.0	
	60h, FC.		112											
			80											
			95											
			107											
C2197-2	1650-1700F for 1h/		56							63.2	87.1	29.0		Supplier Test Report data
	in. WQ; 1200-1250F		44											
	for 1h/in. AC; 1100-		44											
	1150F for 60h, FC.													
C2197-2	As per Table 3.2-3		54				≤+10			69.0	89.2	26.0	67.4	Mt. Vernon Qualification
	plus 1100-1150F		58							70.0	91.5	25.0	65.2	Tests
	for 60h, FC.		65											
			39											
			45											
			26											
C3265-1	1625-1675F for 1h/		44							53.7	85.5	18.0	49.0	Supplier Test Report data
	in. WQ; 1200-1250F		37											
	for 1h/in., AC; 1100-		25											
	1150F for 60h, FC.													
C3265-1	As per Table 3.2.3,		34				≤+10			66.3	87.0	28.1	70.4	Mt. Vernon Qualification
	plus 1100-1150F for		64							66.0	88.8	27.3	69.4	Tests
	40h, FC.		27											
			37											
			65											
			63											
C3278-1	1625-1675F for 1h/in		37							59.4	82.3	29.0	61.6	Supplier Test Report data
	WQ; 1200-1225F for		58								88.5			
	1h/in. AC; 1100-1150F		30											
	for 60h, FC.													

Table 3.2-4. (Cont'd)

Material ID	Heat treatment	Toughness Properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile Properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
C3278-1	As per Table 3.2-3, plus 1100-1150F for 40h, FC.		35 29 53 65 94 60				≤+10			63.5 68.5	84.5 87.2	28.1 29.0	70.4 71.5	Mt. Vernon Qualification Tests
C2800-1	1625-1675F for 1h/in, WQ; 1200-1250F for 1h/in, AC. 1100-1150F, FC.		85 80 66							69.0	92.5	26.0		Supplier Test Report data
C2800-1	As per Table 3.2-3, plus 1100-1150F for 60h, FC.		44 39 36 36 39 39				≤+10			65.5 60.5	89.5 85.0	29.0 29.0	71.5 73.6	Mt. Vernon Qualification Tests
C2800-2	As per Table 3.2-3, plus 1100-1150F for 40h, FC		+20F 43 34 51 32 33 49				0			70.5 69.0	88.5 90.5	28.1 25.0	71.4 71.4	Mt. Vernon Qualification Tests
122S347VA1	As per Table 3.2-3, plus 1125F for 60h, FC.		92 70 70 72 88 47							74.5 72.5	94.5 93.5	24.0 24.5	69.2 70.1	Supplier Test Report data
SA 1135	1100-1150F for 48h, FC (assumed)		56 44 55							67.0 69.5	82.0 81.5	28.0 28.5		Weld Qualification Report data
SA 1229	1100-1150F for 48h, FC (assumed)		55 45 40							72.0 73.2	86.5 87.4	26.5 26.0	65.2 64.3	Weld Qualification Report data

Table 3.2-4. (Cont'd)

Material ID	Heat treatment	Toughness Properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile Properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
WF 25	1100-1150F for 48h, FC		38 28 49							66.5	80.8	27.3		Weld Qualification Report data
SA 1585	1100-1150F for 80h, FC		Surface 31 32 31								Surface 83.0			Weld Qualification Report data
			Center 50 54 51								Center 81.0			
WF 9	1100-1150F for 48h, FC		46 43 45							65.0	81.0	30.5		Weld Qualification Report data
SA 1073	1100-1125F for 8 6h cycles, FC		40 45 39							70.0 74.0	86.5 87.0	26.5 26.0		Weld Qualification Report data
SA 1493	1100-1150F for 48h, FC		41 35 40								82.0			Weld Qualification Report data
SA 1430	1100-1150F for 48h, FC		54 52 53							66.5	84.5	26.0	64.8	Weld Qualification Report data
SA 1426	1100-1150F for 48h, FC		Surface 46 31 36							65.5	82.0	30.0	64.9	Weld Qualification Report data
			Center 35 45 45							67.0	82.0	27.5	66.3	

Figure 3.2-1. The Location and Identification of Materials Used in the Fabrication of the Reactor Pressure Vessel for Oconee Unit 1

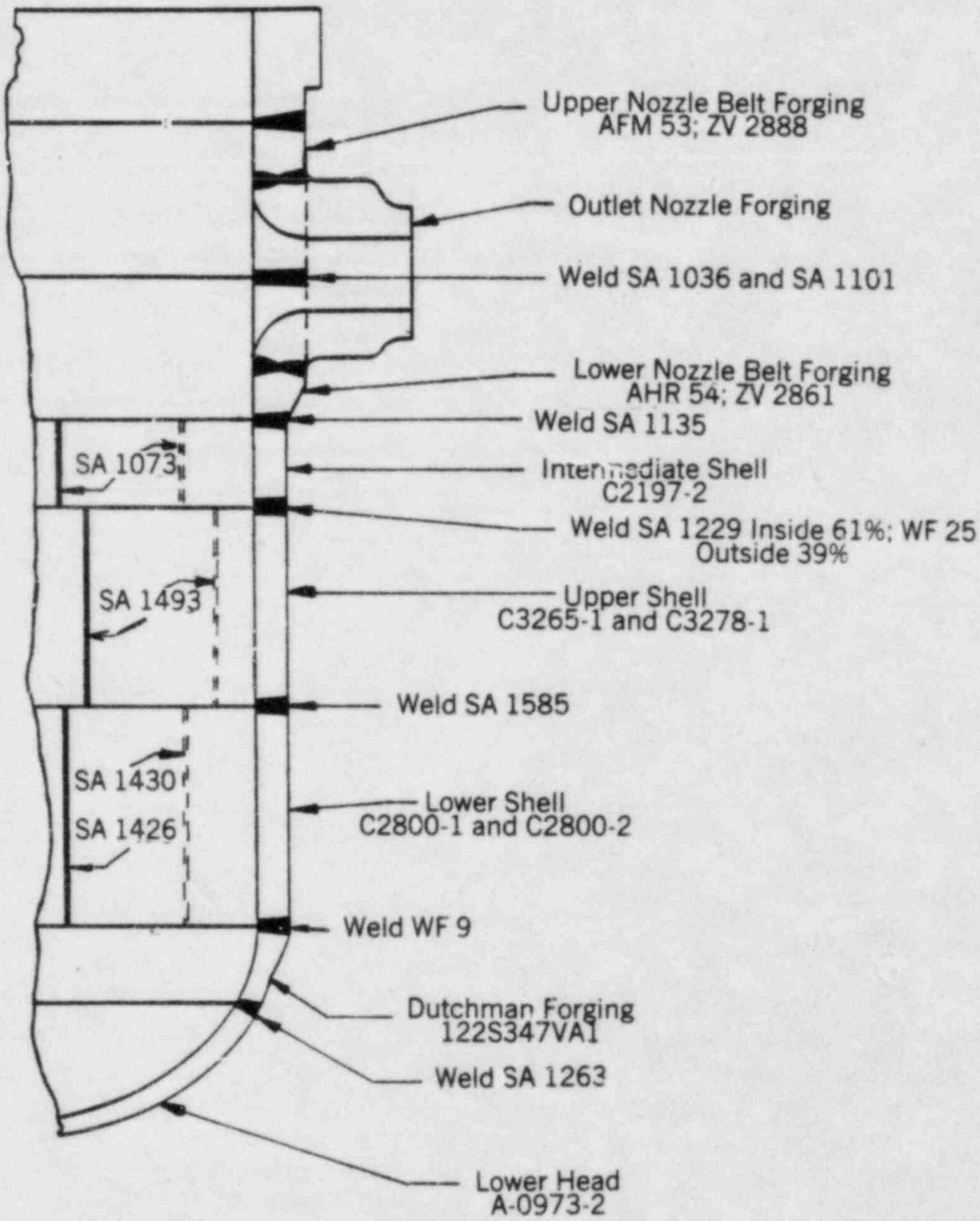
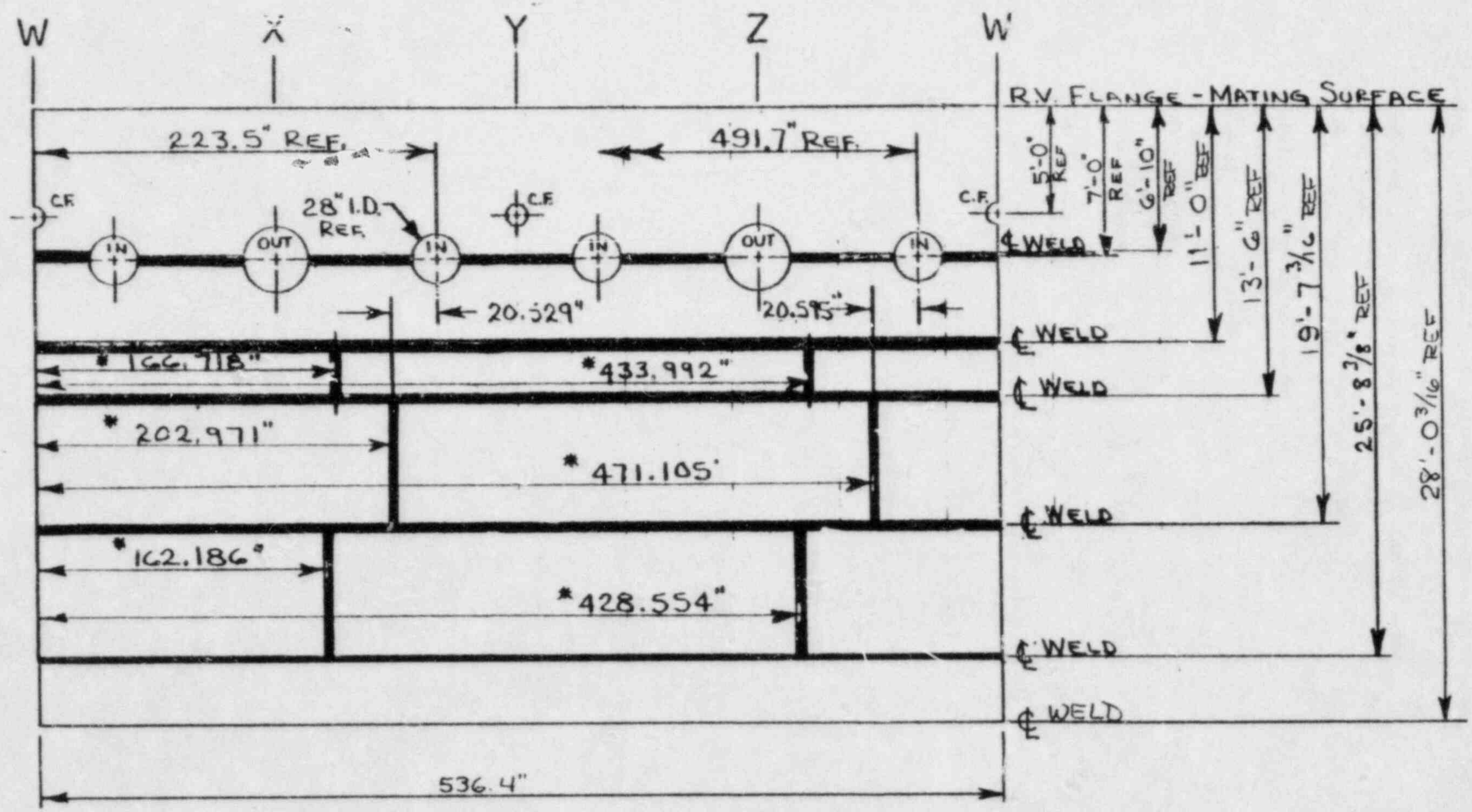
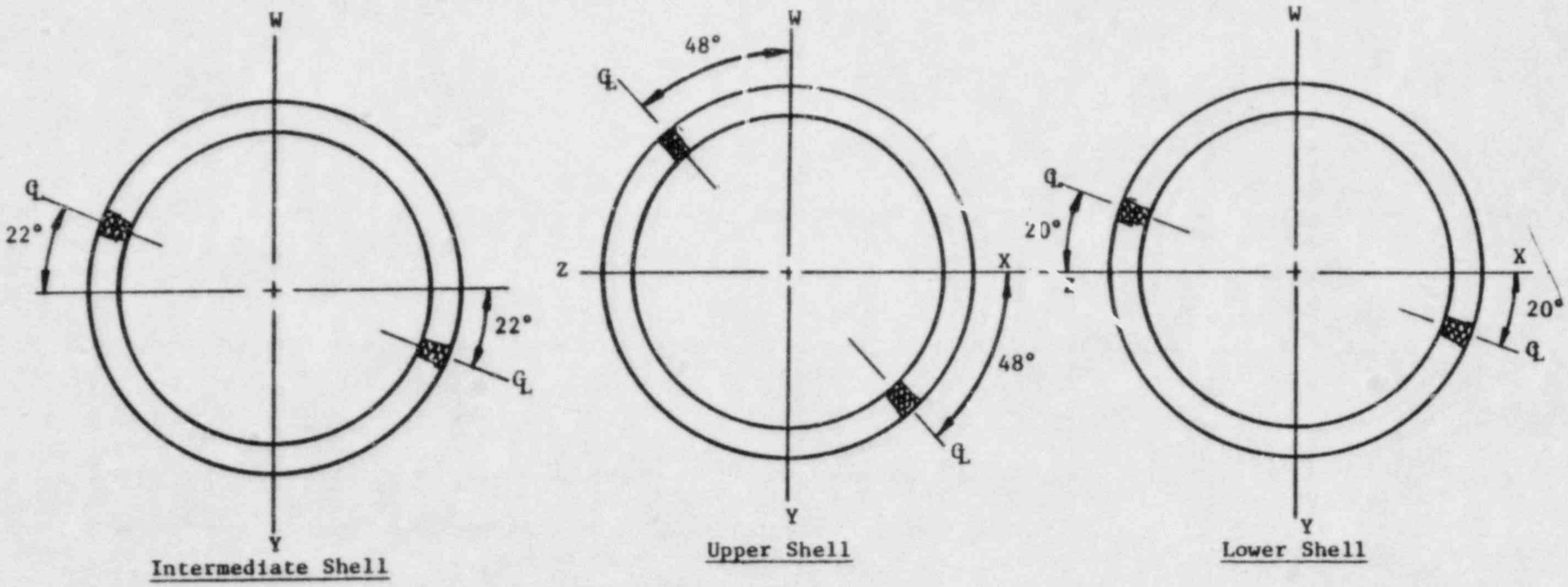


Figure 3.2-2. Reactor Vessel Weldment Locations in Oconee Unit 1



Dimensions based on inside travel

Figure 3.2-3. The Location of Longitudinal Welds in the Upper, Lower, and Intermediate Shell Courses



3.3. Oconee Unit 2 (NSS-4)

Table 3.2-1. General Plant Information for Oconee Unit 2 (NSS-4)

Plant Designation: Oconee Nuclear Station, Unit 2

Plant Location: Seneca, South Carolina

Start-up Date (on power grid): 12/5/73

Owner Utility: Duke Power Company

B&W Contract Number: 620-0004

Reactor Vessel Information

ASME Code Date: 1965 edition including Addenda through Summer 1967

Fabrication Site: Barberton, Ohio and Mt. Vernon, Indiana

Fabrication Schedule:

Start Date -- 1/11/68
Ship Date -- 11/6/70

Vessel Materials:

Nozzle Belt Forgings -- A 508 Cl.2 modified by ASME Code Case 1332-2
Beltline Base Materials -- A 508 Cl.2 modified by ASME Code Case 1339
Beltline Weld Materials -- ASA weld; MnMoNi wire, Linde 80 flux

Location of Materials in Vessel: See Figure 3.3-1.

Chemical Composition of Materials: See Table 3.3-2.

Heat Treatment of Materials: See Table 3.3-3.

Vessel Stress Relief: 1100-1150F for 41h

Table 3.3-2. Chemical Composition of Beltline Region Materials for Oconee Unit 2

	Chemical Composition, wt %									
Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
AMX 77; 123T382	0.25*	0.65	0.006	0.009	0.23	0.76	0.36	0.64	--	0.03 V, 0.01 Co Product analysis
	0.21	0.63	0.006	0.010	0.25	0.72	0.34	0.61	0.06	0.04 V, 0.01 Co Ladle analysis
AAW 163; 3P2359	0.24*	0.63	0.006	0.012	0.25	0.75	0.36	0.62	0.04	0.02 V, 0.01 Co Product analysis
	0.23	0.63	0.006	0.013	0.27	0.78	0.31	0.62	0.04	0.04 V, 0.01 Co Ladle analysis
	0.22	0.66	0.006	0.012	0.27	0.78	0.36	0.58	0.04	0.03 V, 0.01 Co Ladle analysis
AWG 164; 4P1885	0.21*	0.62	0.010	0.010	0.23	0.80	0.39	0.58	0.02	0.01 V, 0.01 Co Product analysis
	0.21	0.66	0.006	0.001	0.27	0.77	0.39	0.61	0.05	0.03 V, 0.01 Co Average of 3 ladle analyses
122T293VA1	0.20#	0.63	0.010	0.011	0.23	0.72	0.38	0.57	--	0.02 V, 0.007 Co Check analysis
	0.20#	0.65	0.010	0.012	0.23	0.71	0.36	0.57	--	0.02 V, 0.009 Co Check analysis
	0.20	0.66	0.007	0.014	0.24	0.70	0.35	0.61	0.11	0.02 V, 0.008 Co Ladle analysis
WF 154	0.07*	1.54	0.013	0.016	0.42	0.59	0.07	0.40	0.31	BAW-1500, Table 10
	0.08	1.48	0.016	0.016	0.53	0.59	0.06	0.40	0.31	BAW-1500, Table 6
	0.072	1.50	0.015	0.021	0.45	0.59	--	0.30	0.20	Weld qualification

Table 3.3-2. (Cont'd)

Material ID	Chemical Composition, wt % C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
WF 25	0.09*	1.60	0.015	0.016	0.50	0.67	0.09	0.42	0.35	BAW-1500, Table 10
	0.09	1.58	0.014	0.016	0.53	0.68	0.09	0.42	0.35	BAW-1500, Table 6
	0.088	1.50	0.019	0.010	0.45	0.71	0.11	0.33	0.29	Weld qualification
WF 112	0.08*	1.47	0.016	0.015	0.54	0.59	0.07	0.40	0.31	BAW-1500, Table 10
	0.08	1.48	0.016	0.016	0.53	0.59	0.06	0.40	0.31	BAW-1500, Table 6
	0.075	1.50	0.024	0.006	0.60	0.58	--	0.51	0.221	Weld qualification

*Analysis to be used for licensing.
#Average of multiple analysis to be used for licensing.

Table 3.3-3. Heat Treatment of Beltline Region Materials for Oconee Unit 2

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
AMX 77 Lower nozzle belt forging	123T382	NA	A 508 Cl.2 mod. by ASME Code Case 1332-2	Ladish	1580F ±20F for 7h, WQ; 1240F ±20F for 14h, WQ.
AAW 163 Upper shell	3P2359	NA	A 508 Cl.2 mod. by ASME Code Case 1332-4	Ladish	1640F ±20F for 4h, WQ; 1590F ±20F for 4h, WQ; 1260F ±20F for 10h, WQ.
AWG 164 Lower shell	4P1885	NA	A 508 Cl.2 mod. by ASME Code Case 1332-4	Ladish	1640F ±10F for 4h, WQ; 1590F ±10F for 4h, WQ; 1260F ±20F for 10h, WQ
122T293VA1 Dutchman forging	122T293	NA	A 508 Cl.2 mod. by ASME Code Case 1332-2	Bethlehem Steel	1550F for 11h, WQ; 1220F for 14h, AC.
WF 154 NB to US circ. weld	406L44	Linde 80 8720		Union Carbide	
WF 25 US to LS circ. weld	299L44	Linde 80 8650		Amer. Chain & Cable	

Table 3.3-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
WF 112 LS to Dutchman circ. weld	406L44	Linde 80 8688		Union Carbide	

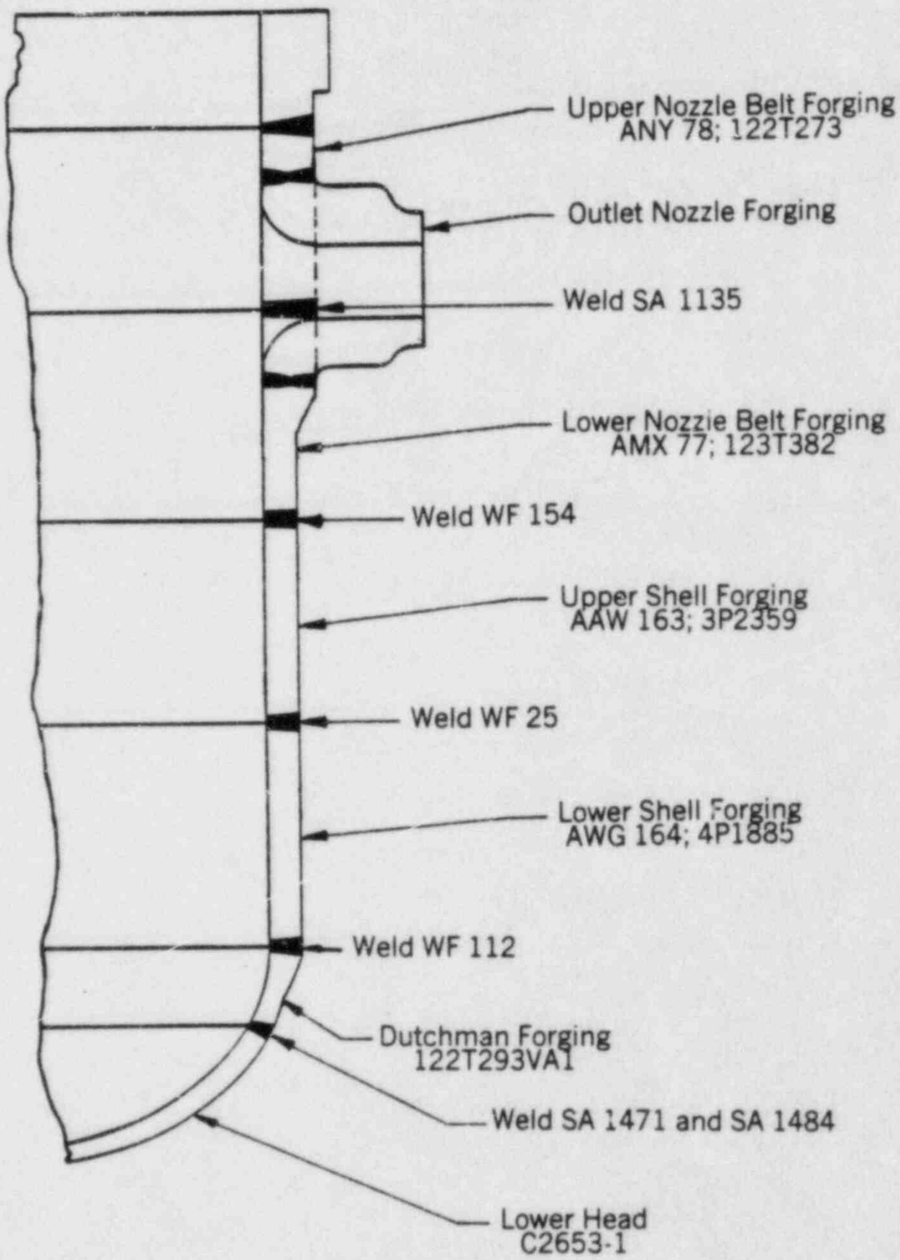
Note: Welds and base metal may have received intermediate stress reliefs. All received full vessel stress relief of 1100-1150F for 41h.

[a] NA denotes not applicable.
[b] FC denotes furnace cool; AC, air cool; WQ, water quench, and BQ, brine quench.

Table 3.3-4. Unirradiated Mechanical Properties of Beltline Region Materials for Oconee Unit 2

Material ID	Heat treatment	Toughness Properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile Properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
AMX 77	As per Table 3.3-3,		90							75.1	94.9	25.0	71.0	Supplier Test Report data
	plus 1125F ±25F for		121							75.4	93.3	25.0	73.0	
	60h, FC.		106											
			103											
			91											
			128											
AAW 163	As per Table 3.3-3,			-55	-25	128	+20	+20		77.0	97.9	24.0	70.0	Supplier Test Report data
	plus 1125F ±25F for													-- three locations tested
	40h, FC			-30	-15	135	+10	+10		73.0	93.7	26.0	72.0	
										74.5	96.2	24.0	70.0	
				-40	-10	140	+20	+20		72.1	93.2	25.0	70.0	
AWG 164	As per Table 3.3-3,			-85	-65	145	+20	+20		72.0	91.9	26.0	72.0	Supplier Test Report data
	plus 1125F ±25F for													-- three locations tested
	40h, FC			-75	-45	145	+20	+20		71.5	91.4	26.0	71.0	
										71.3	90.6	25.0	69.0	
				-75	-50	148	+20	+20		72.5	91.8	25.0	71.0	
122T293VA1	As per Table 3.3-3,		106							75.5	96.5	24.0	68.6	Supplier Test Report data
	plus 1125F ±25F for		101							79.0	99.0	24.5	68.1	
	60h, FC		48											
			82											
			69											
			70											
WF 154	1100-1150F for 48h, FC		41							65.3	81.5	26.6	66.0	Weld Qualification Test data
			37											
			43											
WF 25	1100-1150F for 48h, FC		38							66.5	80.8	27.3		Weld Qualification Test data
			28											
			49											
WF 112	1100-1150F for 48h, FC		35							66.0	83.0	29.7	61.2	Weld Qualification Test data
			40											
			30											

Figure 3.3-1. The Location and Identification of Materials Used in the Fabrication of the Reactor Pressure Vessel for Oconee Unit 2



3.4. Three Mile Island Unit 1 (NSS-5)

Table 3.4-1. General Plant Information for Three Mile Island Unit 1 (NSS-5)

Plant Designation: Three Mile Island Unit 1 (TMI-1)

Plant Location: Londonderry Township, Pennsylvania

Start-up Date (on power grid): 6/19/74

Owner Utility: General Public Utilities (Nuclear)

B&W Contract Number: 620-0005

Reactor Vessel Information

ASME Code Date: 1965 edition including Addenda through Summer 1967

Fabrication Site: Barberton, Ohio and Mt. Vernon, Indiana

Fabrication Schedule:

Start Date -- 1/16/68
Ship Date -- 6/25/70

Vessel Materials:

Nozzle Belt Forgings -- A 508 Cl.2 modified by ASME Code Case 1332-2
Beltline Base Materials -- SA-302 Gr. B modified by ASME Code Case 1339
Beltline Weld Materials -- ASA weld; MnMoNi wire, Linde 80 flux

Location of Materials in Vessel: See Figure 3.4-1.

Chemical Composition of Materials: See Table 3.4-2.

Heat Treatment of Materials: See Table 3.4-3.

Vessel Stress Relief: Upper Shell -- 1100-1150F for 36h (cumulative)
Lower Shell -- 1100-1150F for 37-1/4h (cumulative)

Table 3.4-2. Chemical Composition of Beltline Region Materials for Three Mile Island Unit 1

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
ARY 59; 123S454	0.26*	0.63	0.006	0.008	0.28	0.72	0.34	0.64	0.08	0.04 V, 0.01 Co product analysis
	0.23	0.63	0.010	0.009	0.29	0.69	0.32	0.60		0.04 V, 0.010 Co ladle analysis
C2789-1	0.24	1.36	0.010	0.017	0.23	0.57	0.19	0.51	0.09	0.015 Co
C2789-2	0.24	1.36	0.010	0.017	0.23	0.57	0.19	0.51	0.09	0.015 Co
C3307-1	0.21	1.24	0.010	0.016	0.27	0.55	0.12	0.47	0.12	0.015 Co
C3251-1	0.23	1.41	0.012	0.013	0.21	0.50	0.14	0.47	0.11	0.07 Co Ladle analysis
122T229VA1	0.22#	0.63	0.010	0.008	0.28	0.74	0.38	0.60	--	0.02 V, 0.013 Co Check analysis
	0.22#	0.63	0.010	0.008	0.28	0.72	0.36	0.60	--	0.02 V, 0.013 Co Check analysis
	0.21	0.64	0.008	0.010	0.27	0.69	0.36	0.60	0.08	0.01 V, 0.012 Co Ladle analysis
WF 70	0.09*	1.63	0.018	0.009	0.54	0.59	0.10	0.40	0.35	BAW-1500, Table 10
	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	BAW-1500, Table 6
	0.07	1.60	0.014	0.011	0.48	0.46	--	0.40	0.27	Weld qualification
WF 25	0.09*	1.60	0.015	0.016	0.50	0.68	0.09	0.42	0.35	BAW-1500, Table 10
	0.09	1.58	0.014	0.016	0.53	0.68	0.09	0.42	0.35	BAW-1500, Table 6
	0.088	1.50	0.019	0.010	0.45	0.71	0.11	0.33	0.29	Weld qualification

Table 3.4-2. (Cont'd)

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
	Chemical Composition, wt %									
WF 67	0.08*	1.55	0.021	0.016	0.51	0.60	0.09	0.39	0.24	BAW-1500, Table 10
	0.08	1.55	0.018	0.018	0.52	0.60	0.09	0.39	0.24	BAW-1500, Table 6
	0.064	1.49	0.014	0.017	0.54	0.57	0.02	0.41	0.27	Weld qualification
WF 8	0.06*	1.45	0.010	0.009	0.53	0.55	0.12	0.41	0.29	BAW-1500, Table 10
	0.063	1.45	0.009	0.009	0.53	0.61	--	0.47	0.20	Weld qualification
SA 1526	0.09*	1.53	0.013	0.017	0.53	0.68	0.09	0.42	0.35	BAW-1500, Table 10
	0.09	1.58	0.014	0.016	0.53	0.68	0.09	0.42	0.35	BAW-1500, Table 6
	0.06	1.40	0.016	0.012	0.40	0.60	0.06	0.43	0.46	Weld qualification
SA 1494	0.09*	1.52	0.015	0.012	0.44	0.63	0.08	0.37	0.18	BAW-1500, Table 10
	0.09	1.52	0.015	0.012	0.44	0.45	0.07	0.42	0.14	Weld qualification

*Analysis to be used for licensing.
#Average of multiple analysis to be used for licensing.

Table 3.4-3. Heat Treatment of Beltline Region Materials for Three Mile Island Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
ARY 59 Lower nozzle belt forging	123S454	NA	A 508 Cl.2, mod. by ASME Code Case 1332-2	Ladish	1600F ±20F for 7 h, WQ; 1230F ±20F for 14h, WQ.
C2789-1 Upper shell	C2789-1	NA	SA-302 Gr.B, mod. by ASME Code Case 1339	Lukens	1625-1675F for 1 h/in., WQ; 1200-1250F for 1 h/in., AC. Probably also 1600-1650F for 9-1/2 h, BQ; 1200-1225F for 9-1/2 h, BQ; 1600-1650F for 9-1/2 h, BQ; 1600-1650F for 9-1/2 h, BQ; 1510-1535F for 5 h, BQ; 1200-1225F for 5 h, BQ.
C2789-2 Upper shell	C2789-2	NA	SA-302 Gr.B, mod. by ASME Code Case 1339	Lukens	Supplier heat treatment unknown. Probably also 1600-1650F for 9-1/2 h, BQ; 1200-1225F for 9-1/2 h, BQ; 1600-1650F for 9-1/2 h, BQ; 1600-1650F for 9-1/2 h BQ; 1510-1535F for 5 h, BQ; 1200-1225F for 5 h, BQ.
C3307-1 Lower shell	C3307-1	NA	SA-302 Gr.B, mod. by ASME Code Case 1339	Lukens	Supplier heat treatment unknown. Probably also 1600-1650F for 9-1/2 h, BQ; 1200-1225F for 9-1/2 h, BQ; 1225-1250F for 9-1/2 h, BQ.
C3251-1 Lower shell	C3251-1	NA	SA-302 Gr.B, mod. by ASME Code Case	Lukens	Supplier heat treatment unknown. Probably also 1600-1650F for 9-1/2 h, BQ; 1200-1225F for 9-1/2 h, BQ.
122T229VA1 Dutchman forging	122T229	NA	A 508 Cl.2, mod. by ASME Code Case 1332-2	Bethlehem Steel	1550F for 14 h, WQ; 1220F for 14h, FC.

Table 3.4-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
WF 70 NB to US circ. weld and LS to Dutchman circ. weld	72105	Linde 80 8669	--	Amer. Chain & Cable	
WF 25 US to LS circ. weld	299L44	Linde 80 8650	--	Union Carbide	
WF 67 LS to Dutchman circ. weld	72442	Linde 80 8669	--	Amer. Chain & Cable	
WF 8 US long. weld	8T1762	Linde 80 8532	--	US Steel	
SA 1526 LS long. weld	299L44	Linde 80 8596	--	Union Carbide	
SA 1494 LS long. weld	8T1554	Linde 80 8579	--	US Steel	

(a)NA denotes not applicable.

(b)FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Note: Welds and base metal may have received intermediate stress reliefs. All received full vessel stress relief. Upper shell cumulative received 36 h at 1100-1150F. Lower shell cumulative received 37-1/4 h at 1100-1150F. US to LS circ. weld stress relief is believed to be either 22-1/2 or 27-1/2 h, based on interpretation of RVSP weld records.

Table 3.4-4. Unirradiated Mechanical Properties of Beltline Region Materials for Three Mile Island Unit 1

		Toughness properties							Tensile properties					
Material ID	Heat treatment	Orien-tation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Orien-tation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
ARY 59	As per Table 3.4-3, plus 1125F ±25F for 60 h, FC		117 110 101 120 122 123							69.4 70.2	89.6 92.0	26.0 26.0	73.0 72.0	Supplier Test Report data
C2789-1	1625-1675F for 1 h/in., WQ; 1200-1250F for 1 h/in., AC; 1100-1150F for 60 h, FC		65 67 67							66.5 63.4	90.8 92.5	26 25	 64.5	Supplier Test Report data
C2789-1	As per Table 3.4-3, plus 1100-1150F for 40 h, FC		50 36 33				0			71.0 69.5	94.0 94.0	22.7 25.8	62.2 63.5	Mt. Vernon Qualification Test data
C2789-2	As per Table 3.4-3, plus 1100-1150F for 40 h, FC		42 37 35				+10			70.0 67.0	93.8 92.0	23.4 26.6	62.5 63.5	Mt. Vernon Qualification Test data
C3307-1	As per Table 3.4-3, plus 1100-1150F for 40 h, FC		+30F 40 56 51 42 41 29				+10			64.5 64.2	86.5 86.0	32.8 31.3	71.4 70.4	Mt. Vernon Qualification Test data
C3251-1	As per Table 3.4-3, plus 1100-1150F for 40 h, FC		43 40 29 71 59 26				-10			66.8 70.0	87.5 88.8	28.0 25.7	71.4 70.4	Mt. Vernon Qualification Test data

Table 3.4-4. (Cont'd)

		Toughness properties							Tensile properties					
Material ID	Heat treatment	Orien-tation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Orien-tation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
122T229VA1	As per Table 3.4-3, plus 1125F for 60 h, FC		84							75.5	96.5	24.5	69.5	Supplier Test Report data
			45							74.0	95.0	25.0	69.7	
			100											
			116											
			110											
			117											
WF 70	1100-1150F for 48 h, FC		39							69.0	85.5	25.8	64.7	Weld Qualification Test data
			35											
			44											
WF 25	1100-1150F for 48 h, FC		38							66.5	80.8	27.3		Weld Qualification Test data
			28											
			49											
WF 67	1100-1150F for 48 h, FC		29							64.0	81.5	31.3	65.8	Weld Qualification Test data
			35											
			30											
WF 3	1100-1150F for 48 h		45							71.0	85.5	25.0		Weld Qualification Test data
			38											
			30											
SA 1526	1100-1150F for 48 h		33								88.0			Weld Qualification Test data
			33											
			33											
SA 1494	1100-1150F for 48 h		54								81.0			Weld qualification Test data
			25											
			44											

Figure 3.4-1. The Location and Identification of Materials Used in the Fabrication of the Reactor Pressure Vessel for Three Mile Island Unit 1

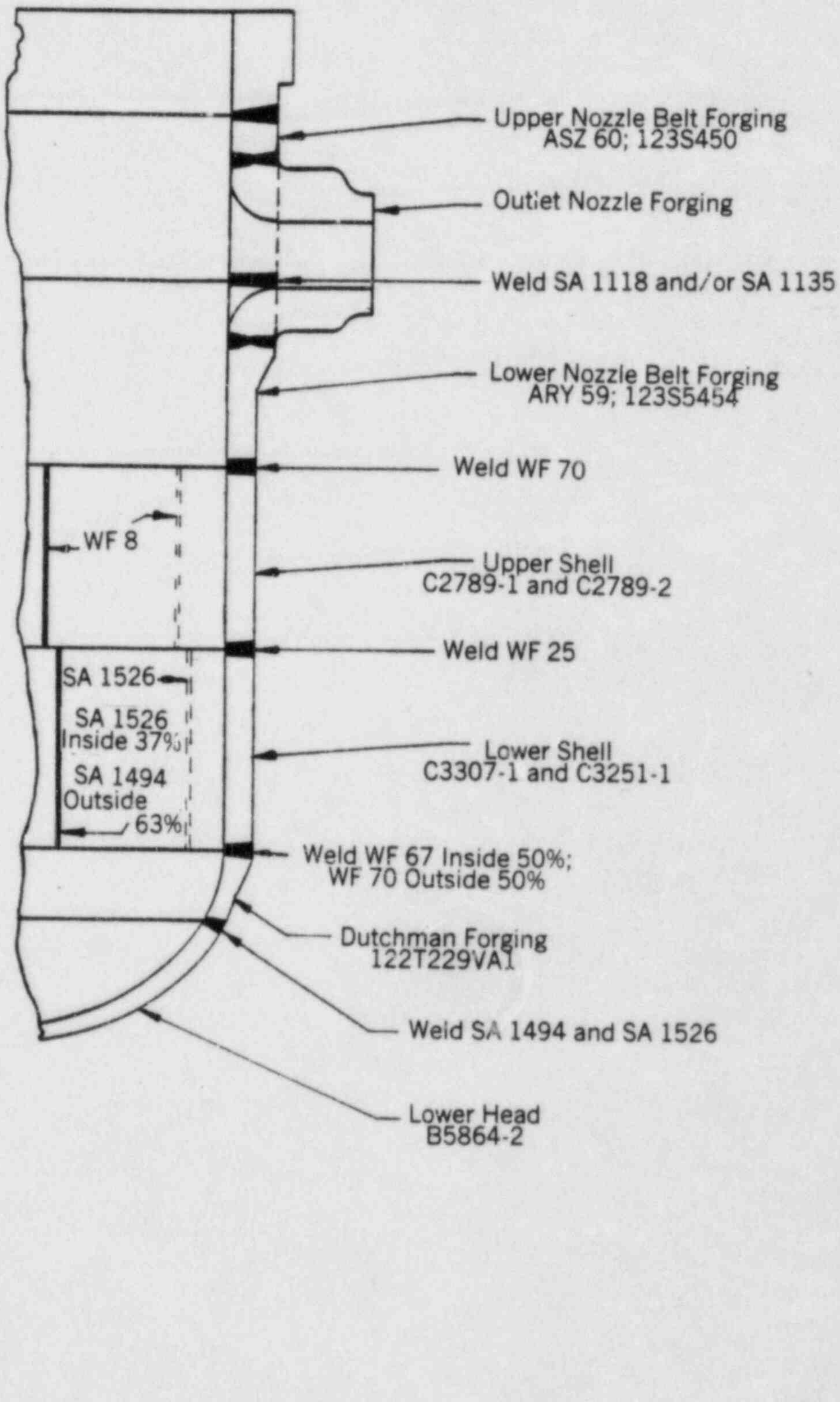
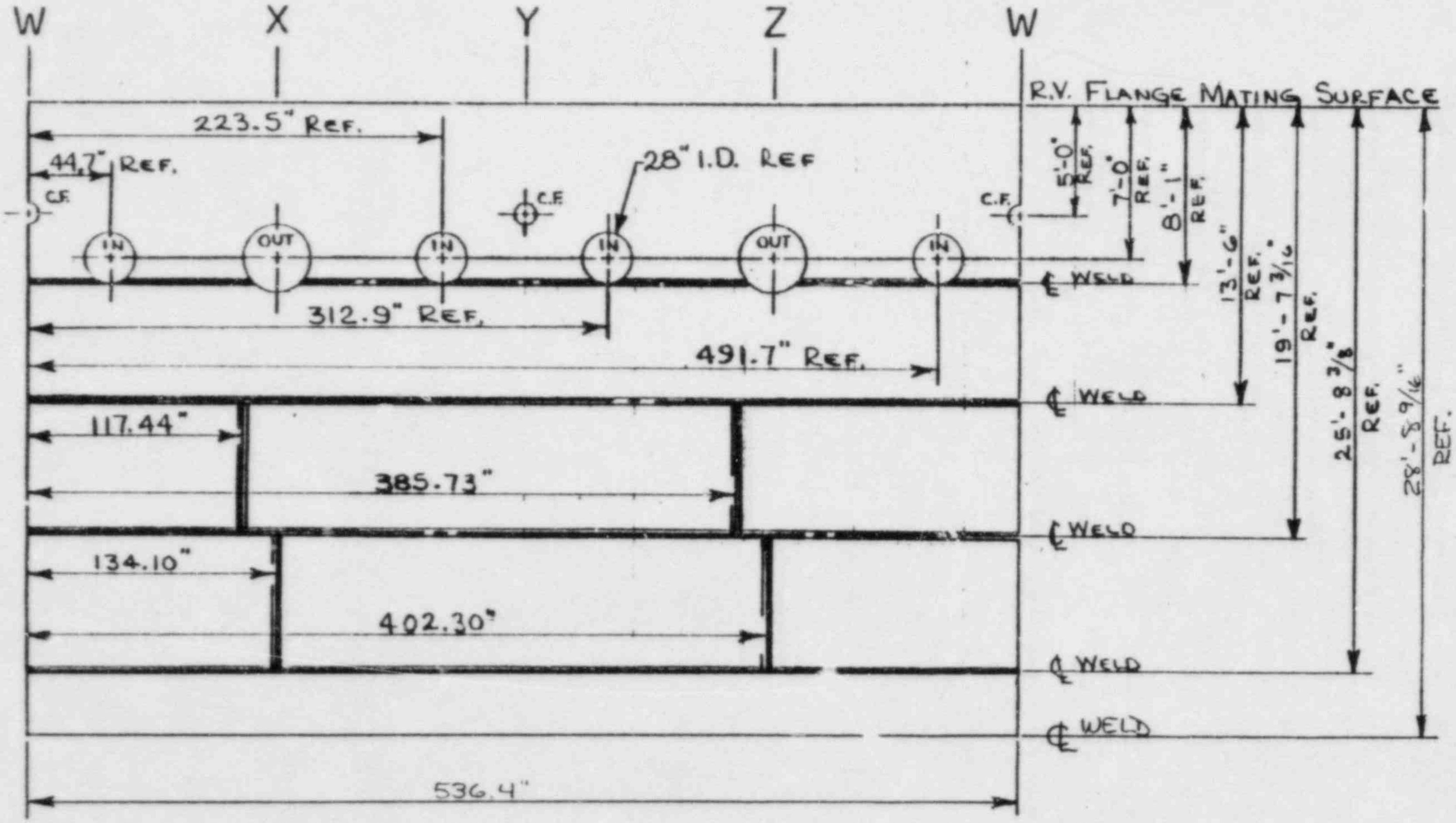
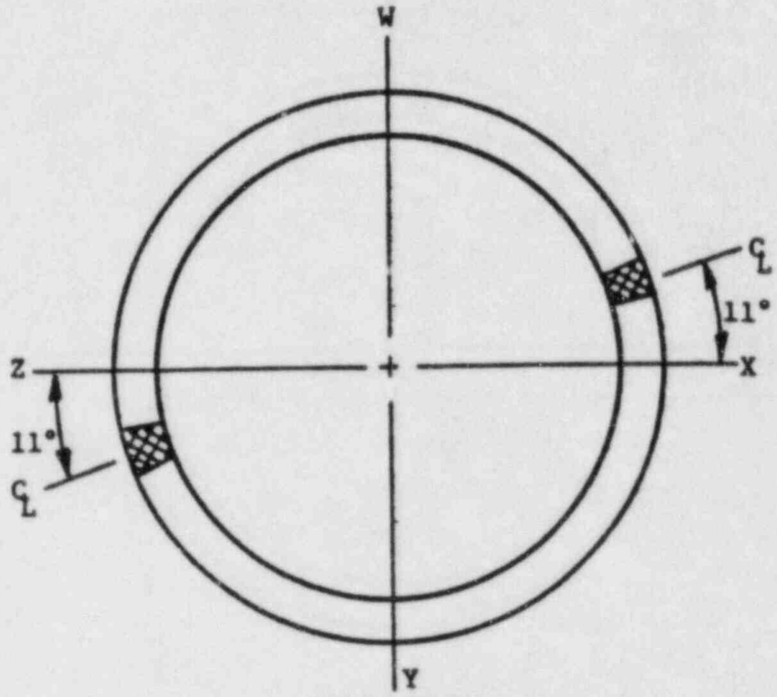


Figure 3.4-2. Reactor Vessel Weldment Locations in Three Mile Island Unit 1

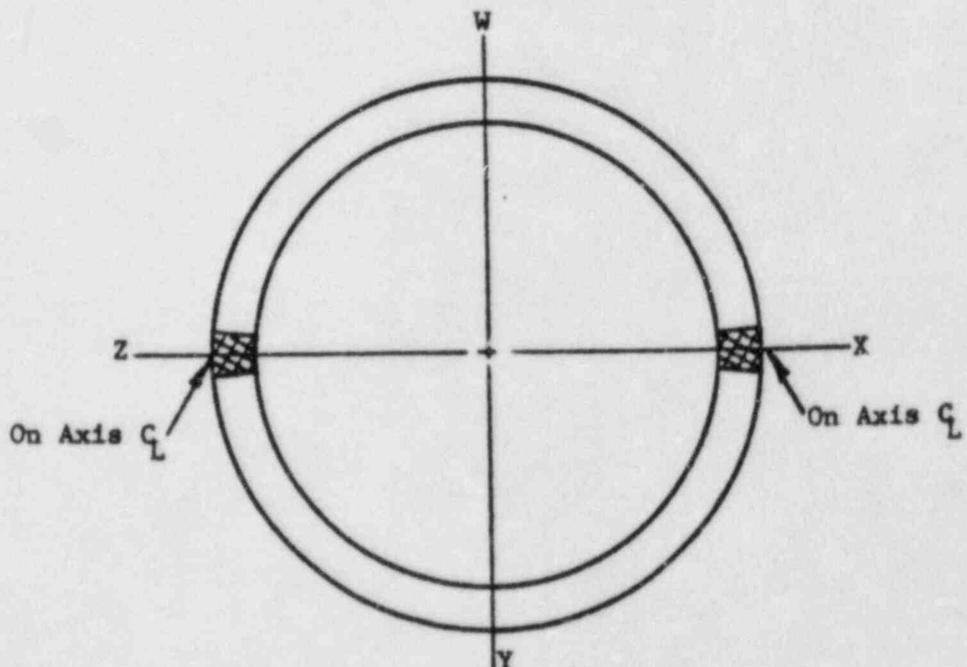


Dimensions are based on inside travel

Figure 3.4-3. The Location of Longitudinal Welds in the Upper and Lower Shell Courses in Three Mile Island Unit 1



Upper Shell



Lower Shell

3.5. Crystal River Unit 3 (NSS-7)

Table 3.5-1. General Plant Information for Crystal River Unit 3 (NSS-7)

Plant Designation: Crystal River Unit 3

Plant Location: Red Level, Florida

Start-up Date (on power grid): 1/30/77

Owner Utility: Florida Power Corporation

B&W Contract Number: 620-0007

Reactor Vessel Information

ASME Code Date: 1965 edition including Addenda through Summer 1967

Fabrication Site: Barberton, Ohio and Mt. Vernon, Indiana

Fabrication Schedule:

Start Date -- 11/15/68
Ship Date -- 6/21/71

Vessel Materials:

Nozzle Belt Forgings -- A 508 Cl. 2 modified by ASME Code Case 1332-2
Beltline Base Materials -- SA-533 Gr. B, Cl. 1
Beltline Weld Materials -- ASA weld; MnMoNi wire, Linde 80 flux

Location of Materials in Vessel: See Figure 3.5-1.

Chemical Composition of Materials: See Table 3.5-2.

Heat Treatment of Materials: See Table 3.5-3.

Vessel Stress Relief: Upper Shell -- 1100-1150F for 27h (cumulative)
Lower Shell -- 1100-1150F for 24-3/4h (cumulative)

Table 3.5-2. Chemical Composition of Beltline Region Materials for Crystal River Unit 3

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
AZJ 94; 123V190	0.26*	0.65	0.007	0.016	0.24	0.72	0.34	0.62	--	0.04 V, 0.02 Co Product analysis
	0.24	0.66	0.009	0.008	0.24	0.71	0.33	0.60	--	0.04 V, 0.012 Co Ladle analysis
C4344-1	0.23	1.30	0.008	0.016	0.22	0.54	0.11	0.55	0.20	0.013 Co
C4344-2	0.23	1.30	0.008	0.016	0.22	0.54	0.11	0.55	0.20	0.013 Co
C4347-1	0.22	1.32	0.013	0.015	0.24	0.58	0.11	0.55	0.12	0.020 Co
C4347-2	0.22	1.32	0.013	0.015	0.24	0.58	0.11	0.55	0.12	0.020 Co
124W295VA1	0.22#	0.59	0.01	0.012	0.24	0.80	0.34	0.59	0.10	0.02 V, 0.014 Co Check analysis
	0.22#	0.58	0.01	0.012	0.23	0.76	0.34	0.57	0.09	0.02 V, 0.011 Co Check analysis
	0.22	0.60	0.010	0.012	0.24	0.76	0.32	0.59	0.08	0.01 V, 0.01 Co Ladle analysis
SA 1769	0.09*	1.49	0.020	0.014	0.56	0.61	0.16	0.37	0.26	BAW-1500, Table 10
	0.08	1.51	0.020	0.013	0.57	0.61	0.16	0.37	0.25	BAW-1500, Table 6
	0.10	1.36	0.021	0.016	0.45	0.66	0.18	0.39	0.19	Weld qualification
WF 169-1	0.08*	1.56	0.016	0.016	0.45	0.63	0.08	0.37	0.18	BAW-1500, Table 10
	0.075	1.58	0.014	0.013	0.45	0.59	0.02	0.42	0.106	Weld qualification

Table 3.5-2. (Cont'd)

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
	Chemical Composition, wt %									
WF 8	0.06*	1.45	0.010	0.009	0.53	0.55	0.12	0.41	0.29	BAW-1500, Table 10
	0.063	1.45	0.009	0.009	0.53	0.61	--	0.47	0.20	Weld qualification
WF 1[illegible]	0.09*	1.45	0.010	0.017	0.39	0.55	0.12	0.41	0.29	BAW-1500, Table 10
	0.091	1.45	0.004	0.017	0.39	0.45	--	0.32	0.105	0.044 Co -- weld qualification
WF 70	0.09*	1.63	0.018	0.009	0.54	0.59	0.10	0.40	0.35	BAW-1500, Table 10
	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	BAW-1500, Table 6
	0.070	1.60	0.014	0.011	0.48	0.46	--	0.40	0.27	Weld qualification
SA 1580	0.07*	1.45	0.015	0.013	0.43	0.55	0.12	0.41	0.29	BAW-1500, Table 10
	0.07	1.45	0.015	0.013	0.43	0.60	0.13	0.43	0.22	Weld qualification
WF 154	0.07*	1.54	0.013	0.016	0.42	0.59	0.07	0.40	0.31	BAW-1500, Table 10
	0.08	1.48	0.016	0.016	0.53	0.59	0.06	0.40	0.31	BAW-1500, Table 6

*Analysis to be used for licensing.
#Average of multiple analysis to be used for licensing.

Table 3.5-3. Heat Treatment of Beltline Region Materials for Crystal River Unit 3

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
AZJ 94 Lower nozzle belt forging	123V190	NA	A 508 Cl. 2, mod. by ASME Code Case 1332-2	Ladish	1590F ±20F for 7 h, WQ; 1270F ±20F for 14 h, WQ.
C4344-1 Upper shell	C4344-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1180F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1175-1200F for 6 h, BQ.
C4344-2 Upper shell	C4344-2	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1100F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1175-1200F for 6 h, BQ.
C4347-1 Lower shell	C4347-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1180F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1250-1275F for 5 h, BQ.
C4347-2 Lower shell	C4347-2	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1180F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1275F for 5 h, BQ.
124W295VA1 Dutchman forging	124W295	NA	A 508 Cl. 2, mod. by ASME Code Case 1332-3	Bethlehem Steel	1550F for 11 h, WQ; 1220F for 4 h, AC.
SA 1769 NB to US circ. weld	71249	Linde 80 8738		Amer. Chain & Cable	

Table 3.5-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
WF 169-1 NB to US circ. weld	8T1554	Linde 80 8754		US Steel	
WF 8 US long. weld	8T1762	Linde 80 8632		US Steel	
WF 18 US long. weld	8T1762	Linde 80 8650		US Steel	
WF 70 US to LS circ. weld	72105	Linde 80 8669		Amer. Chain & Cable	
SA 1580 LS long. weld	8T1762	Linde 80 8596		US Steel	
WF 154 LS to Dutchman circ. weld	406L44	Linde 80 8720		Union Carbide	

(a)NA denotes not applicable.

(b)FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Note: Welds and base metal may have received intermediate stress reliefs. All received full vessel stress relief. Upper shell cumulative received 27 h at 1100-1150F. Lower shell cumulative received 24-3/4 h at 1100-1150F. US to LS circ. weld stress relief is believed to be 27 h, based on RVSP weld records.

Table 3.5-4. Unirradiated Mechanical Properties of Beltline Region Materials for Crystal River Unit 3

		Toughness properties							Tensile properties					
Material ID	Heat treatment	Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
AZJ 94	As per Table 3.5-3, plus 1125F ±25F for 60 h, FC		103 96 97 101 111 91							66.8 67.0	88.2 89.0	27 27	73 74	Supplier Test Report data
C4344-1	1650-1700F for 1 h/in., WQ; 1180F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		39 40 36							62.8	86.7	26		Supplier Test Report data
C4344-1	As per Table 3.5-3, plus 1100-1150F for 40 h, FC	1/4T location		+6	+44	123	-10	-10	1/4T location	71.5 70.0	92.5 90.8	20.3 25.8	59.8 64.7	Mt. Vernon Qualification Test data
C4344-1	As per Table 3.5-3, plus 1100-1150F for 27 h, FC	TL		+50	+80	88	-10	+20						Supplementary Mt. Vernon tests of surveillance material
C4344-2	1650-1700F for 1 h/in., WQ; 1100F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		42 40 30							66.4	91.2	28.0		Supplier Test Report data
C4344-2	As per Table 3.5-3, plus 1100-1150F for 40 h, FC	1/4T location		-2	+31	131	-10	-10	1/4T location	71.0 70.5	92.0 91.8	24.2 25.0	61.1 63.3	Mt. Vernon Qualification Test data
C4344-2	As per Table 3.5-3, plus 1100-1150F for 27 h, FC	TL		+30	+80	88	-10	+20						Supplementary Mt. Vernon tests of surveillance material

Table 3.5-4. (Cont'd)

		Toughness properties							Tensile properties					
Material ID	Heat treatment	Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
C4347-1	1650-1700F for 1 h/in., WQ; 1180F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		53 54 47							68.9	95.1	24.0		Supplier Test Report data
	As per Table 3.5-3, plus 1100-1150F for 40 h, FC	1/4T location		-18	+17	136	-20	-20	1/4T location	66.5	90.0	27.5	64.7	Mt. Vernon Qualification
	As per Table 3.5-3; probably plus 1100-1150F for 27 h, FC	TL		+20	+50	119	-20	-10						Supplementary Mt. Vernon tests of excess shell course material
C4347-2	1650-1700F for 1 h/in., WQ; 1180F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		43 53 63							71.4	95.6	24.0		Supplier Test Report data
	As per Table 3.5-3, plus 1100-1150F for [illegible] h, FC	1/4T location		-19	+16	131	-20	-20	1/4T location	67.0	89.5	27.5	63.5	Mt. Vernon Qualification
	As per Table 3.5-3; probably plus 1100-1150F for 27 h, FC	TL		+85	+105	86	-20	+45						Supplementary Mt. Vernon tests of excess shell course material
124W295VA1	As per Table 3.5-3, plus 1125F for 60 h, FC		82 71 63 51 78 65 80							75.0 73.5	96.5 96.0	25.0 25.0	69.2 69.2	Supplier Test Report data

Table 3.5-4. (Cont'd)

Material ID	Heat treatment	Toughness properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
SA 1769	1100-1150F for 8 6-h cycles, FC		36 35 38								91.0			Weld Qualification Test data
WF 169-1	1100-1150F for 48 h, FC		36 43 42							68.0	83.5	26.6	64.7	Weld Qualification Test data
	1100-1150F for 8 6-h cycles, FC		42 29 46							66.9	82.5	26.6	66.0	Weld Qualification Test data
WF 8	1100-1150F for 48 h		45 38 30							71.0	85.5	25.0		Weld Qualification Test data
WF 18	1100-1150F for 48 h		45 46 38							68.0	84.3	28.1	62.3	Weld Qualification Test data
WF 70	1100-1150F for 48 h		39 35 44							69.0	85.5	25.8	64.7	Weld Qualification Test data
SA 1580	1100-1150F for 80 h		31 29 25 49 41 40								89.0 86.0			Weld Qualification Test data
WF 154	1100-1150F for 48 h		41 37 43							65.3	81.5	26.6	66.0	Weld Qualification Test data

Figure 3.5-1. The Location and Identification of Materials Used in the Fabrication of the Reactor Pressure Vessel for Crystal River Unit 3

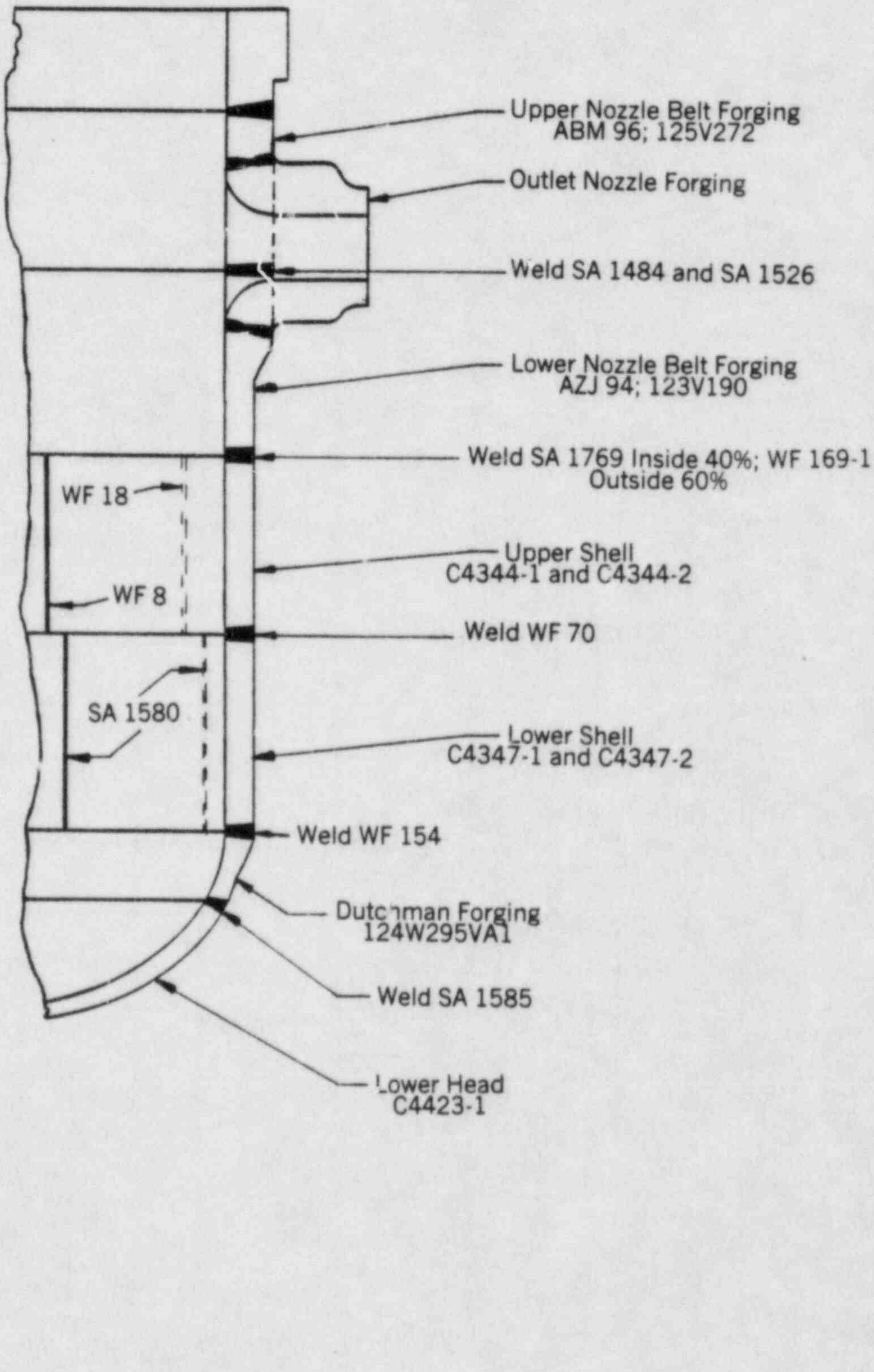
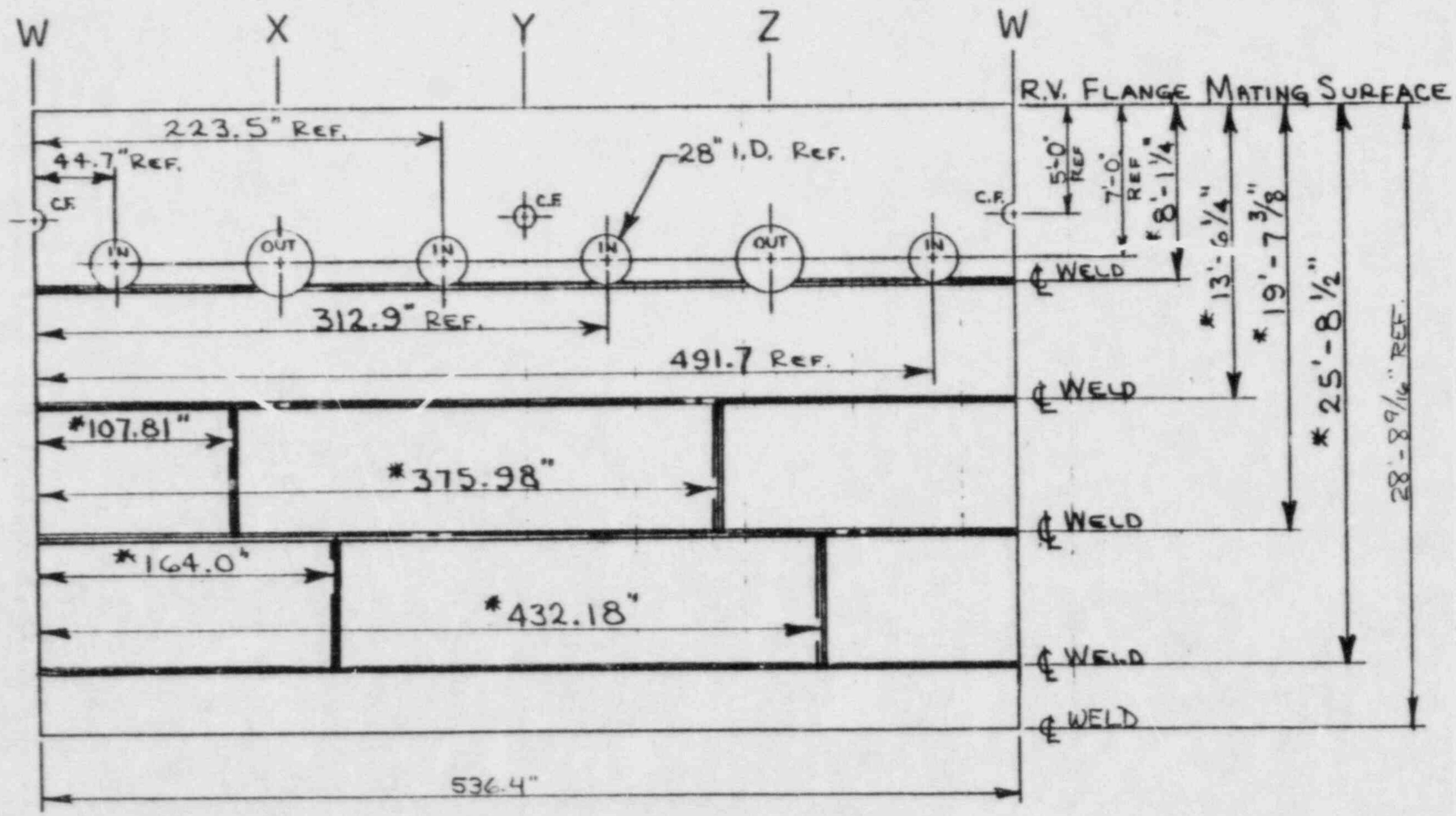


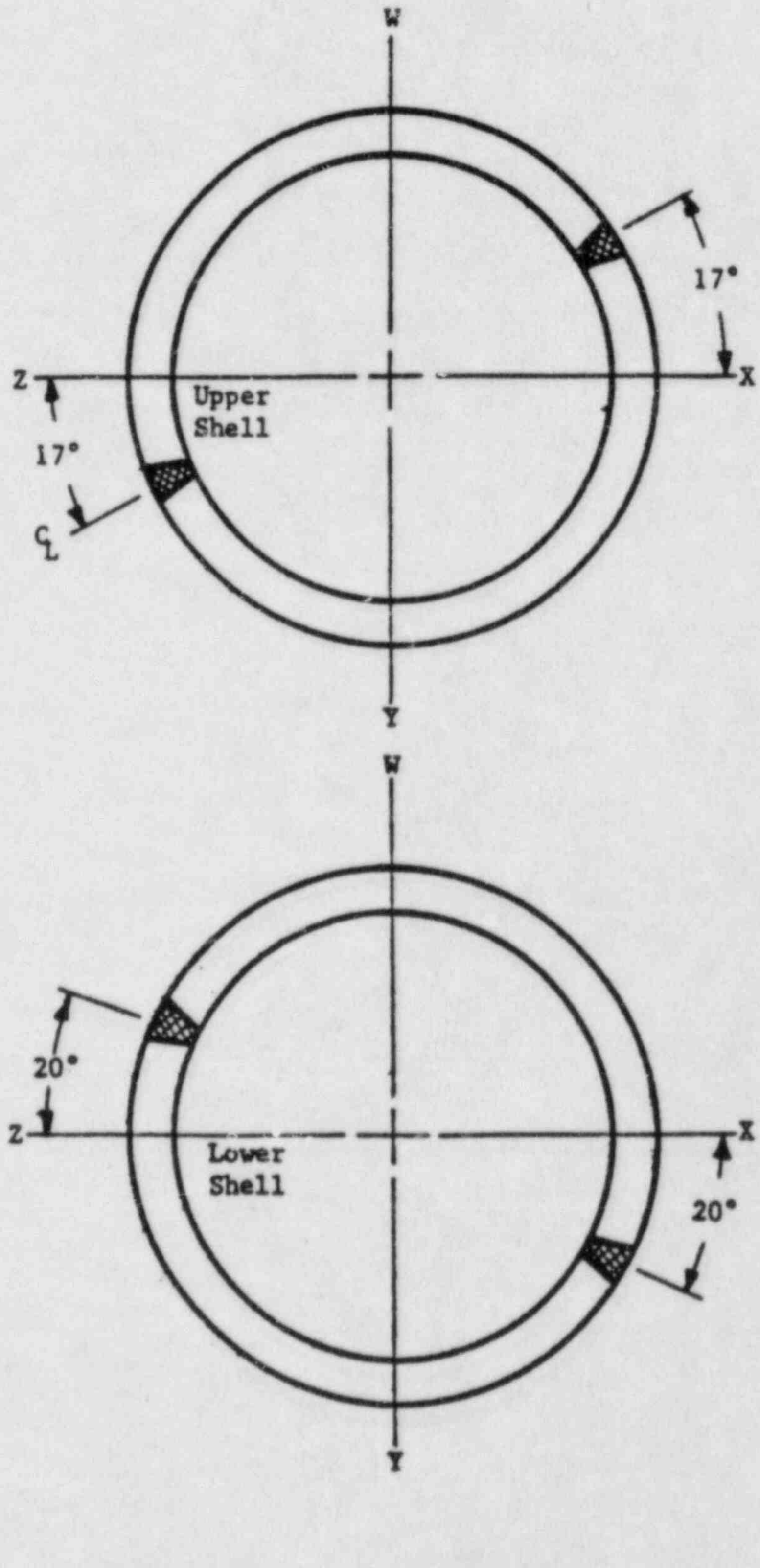
Figure 3.5-2. Reactor Vessel Weldment Locations in Crystal River Unit 3



Dimensions are based on inside travel

*From as-built drawing

Figure 3.5-3. The Location of Longitudinal Welds in the Upper and Lower Shell Courses in Crystal River Unit 3



3.6. Arkansas Nuclear One Unit 1 (NSS-8)

Table 3.6-1. General Plant Information for Arkansas Nuclear One Unit 1 (NSS-8)

Plant Designation: Arkansas Nuclear One Unit 1 (ANO-1)

Plant Location: Russellville, Arkansas

Start-up Date (on power grid): 8/17/74

Owner Utility: Arkansas Power & Light Company

B&W Contract Number: 620-0008

Reactor Vessel Information

ASME Code Date: 1965 edition including Addenda through Summer 1967

Fabrication Site: Barberton, Ohio and Mt. Vernon, Indiana

Fabrication Schedule:

Start Date -- 3/18/69
Ship Date -- 8/30/71

Vessel Materials:

Nozzle Belt Forgings -- A 508 Cl. 2 modified by ASME Code Case 1332-2
Beltline Base Materials -- SA-533 Gr. B, Cl. 1
Beltline Weld Materials -- ASA weld; MnMoNi wire, Linde 80 flux

Location of Materials in Vessel: See Figure 3.6-1.

Chemical Composition of Materials: See Table 3.6-2.

Heat Treatment of Materials: See Table 3.6-3.

Vessel Stress Relief: Upper Shell -- 1100-1150F for 28-1/2 h (cumulative)
Lower Shell -- 1100-1150F for 25-1/4 h (cumulative)

Table 3.6-2. Chemical Composition of Beltline Region Materials for Arkansas Nuclear One Unit 1

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
	Chemical Composition, wt %									
AYN 131; 528360	0.27*	0.64	0.009	0.015	0.21	0.70	0.32	0.66	0.03	0.03 V, 0.02 Co Product analysis
	0.25	0.58	0.013	0.020	0.18	0.66	0.30	0.58	0.05	0.02 V, 0.005 Co Ladle analysis
C5120-2	0.22	1.41	0.014	0.013	0.18	0.55	0.18	0.53	0.17	0.012 Co
C5114-2	0.21	1.32	0.010	0.016	0.20	0.52	0.19	0.57	0.15	0.012 Co
C5120-1	0.22	1.41	0.014	0.013	0.18	0.55	0.18	0.53	0.17	0.012 Co
C5114-1	0.21	1.32	0.010	0.016	0.20	0.52	0.19	0.57	0.15	0.012 Co
125W609VA1	0.23#	0.67	0.01	0.010	0.29	0.76	0.36	0.59	--	0.02 V, 0.010 Co Check analysis
	0.24#	0.56	0.01	0.013	0.27	0.74	0.36	0.59	--	0.02 V, 0.010 Co Check analysis
	0.24	0.66	0.011	0.009	0.27	0.76	0.34	0.62	0.11	0.01 V, 0.017 Co Ladle analysis
WF 182-1	0.08*	1.69	0.014	0.013	0.45	0.63	0.14	0.40	0.24	BAW-1500, Table 10
	0.08	1.69	0.014	0.013	0.45	0.63	0.14	0.40	0.24	BAW-1500, Table 6
	0.071	1.60	0.014	0.015	0.30	0.59	--	0.42	0.18	Weld qualification
	0.09	1.74	0.013	0.017	0.45	0.61	0.14	0.46	0.21	Revised weld qualification

Table 3.6-2. (Cont'd)

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
	Chemical Composition, wt %									
WF 112	0.08*	1.47	0.016	0.015	0.54	0.59	0.07	0.40	0.31	BAW-1500, Table 10
	0.08	1.48	0.016	0.016	0.53	0.59	0.06	0.40	0.31	BAW-1500, Table 6
	0.075	1.50	0.024	0.006	0.60	0.58	--	0.51	0.221	Weld qualification
SA 1788	0.08*	1.43	0.017	0.012	0.42	0.54	0.08	0.38	0.25	BAW-1500, Table 10
	0.08	1.44	0.012	0.011	0.51	0.54	0.08	0.38	0.25	BAW-1500, Table 6
	0.08	1.43	0.017	0.012	0.42	0.47	0.06	0.40	0.29	Weld qualification
WF 18	0.09*	1.45	0.010	0.017	0.39	0.55	0.12	0.41	0.29	BAW-1500, Table 10
	0.091	1.45	0.004	0.017	0.39	0.45	--	0.32	0.105	0.044 Co, weld qualification

*Analysis to be used for licensing.
#Average of multiple analysis to be used for licensing.

Table 3.6-3. Heat Treatment of Beltline Region Materials for Arkansas Nuclear One Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
AYN 131 Lower nozzle belt forging	528360	NA	A 508 Cl. 2 mod. by ASME Code Case 1332-2	Ladish	1580F ±20F for 5 h, WQ; 1250F ±20F for 14 h, WQ.
C5120-2 Upper shell	C5120-2	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ.
C5114-2 Upper shell	C5114-2	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ.
C5120-1 Lower shell	C5120-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ.
C5114-1 Lower shell	C5114-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ.
125W609VA1 Dutchman forging	125W609	NA	A 508 Cl. 2 mod. by ASME Code Case 1332-3	Bethlehem Steel	1550F for 11-1/2 h, WQ; 1230F for 14 h, AC.
WF 182-1 NB to US circ. weld	821T44	Linde 80 8754		Union Carbide Linde	

Table 3.6-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
WF 112 US to LS circ. weld	406L44	Linde 80 8688		Union Carbide	
SA 1788 LS to Dutchman circ. weld	61782	Linde 80 8754		Amer. Chain & Cable	
WF 18 US and LS long. welds	8T1762	Linde 80 8650		US Steel	

[a]NA denotes not applicable.

[b]FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Note: Welds and base metal may have received intermediate stress reliefs. All received full vessel stress relief. Upper shell cumulative received 28-1/2 h at 1100-1150F. Lower shell cumulative received 25-1/4 h at 1100-1150F. US to LS circ. weld stress relief is believed to be 29 h based on RVSP weld records.

Table 3.6-4. Unirradiated Mechanical Properties of Beltline Region Materials for Arkansas Nuclear One Unit 1

Material ID	Heat treatment	Toughness properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
AYN 131	1580F ±20F for 5 h, WQ; 1250F ±20F for 14 h, WQ; 1125F ±25F for 60 h, FC		74 83 62 42 58 69							70.1 76.5	91.1 95.8	26.0 24.0	69.0 67.[illegible]	Supplier Test Report data
C5120-2	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		55 53 49							72.2	92.5	25.0		Supplier Test Report data
C5120-2	As per Table 3.6-3, plus 1100-1150F for 40 h, FC	1/4T location		-16	+12	133	-10	-10	1/4T location	71.0 71.0	91.0 90.8	25.0 19.5	63.4 60.7	Mt. Vernon Qualification Test data
C5114-2	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		40 50 36							65.7	91.1	26.0		Supplier Test Report data
	As per Table 3.6-3, plus 1100-1150F for 40 h, FC	1/4T location		+14	+42	133	-10	-10	1/4T location	72.0 70.8	92.8 90.5	24.2 25.8	61.6 63.7	Mt. Vernon Qualification Test data
C5120-1	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC; 1100-1150 for 60 h, FC		56 48 54							75.0	94.4	[illegible]4.0		Supplier Test Report data
	As per Table 3.6-3, plus 1100-1150F for 40 h, FC	1/4T location		-6	+19	123	-10	-10	1/4T location	70.5 72.0	90.2 94.0	25.8 25.8	61.7 63.2	Mt. Vernon Qualification Test data

Table 3.6-4. (Cont'd)

Material ID	Heat treatment	Toughness properties: Orien-tation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile properties: Orien-tation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
C5114-1	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		57 40 57							69.0	93.0	27.0		Supplier Test Report data
C5114-1	As per Table 3.6-3, plus 1100-1150F for 40 h, FC	1/4T loca-tion		+10	+40	129	0	0	1/4T loca-tion	73.5 71.5	[illegible]4.8 94.2	25.8 22.7	64.4 61.2	Mt. Vernon Qualification Test data
125W609VA1	As per Table 3.6-3, plus 1125F for 60 h, FC		61 95 66 70 61 63							75.5 75.0	99.5 98.5	24.5 24.5	70.3 69.5	Supplier Test Report data
WF 182-1	1100-1150F for 48 h, FC		36 33 44								81.0			Weld Qualification Test data
WF 112	1100-1150F for 48 h, FC		35 40 30							66.0	83.0	29.7	61.2	Weld Qualification Test data
SA 1788	1100-1150F for 8 6 h cycles, FC		40 38 36								81.5			Weld Qualification Test data
WF 18	1100-1150F for 48 h, FC		45 46 38							68.0	84.3	28.1	62.3	Weld Qualification Test data

Figure 3.6-1. The Location and Identification of Materials Used in the Fabrication of the Reactor Pressure Vessel for Arkansas Nuclear One Unit 1

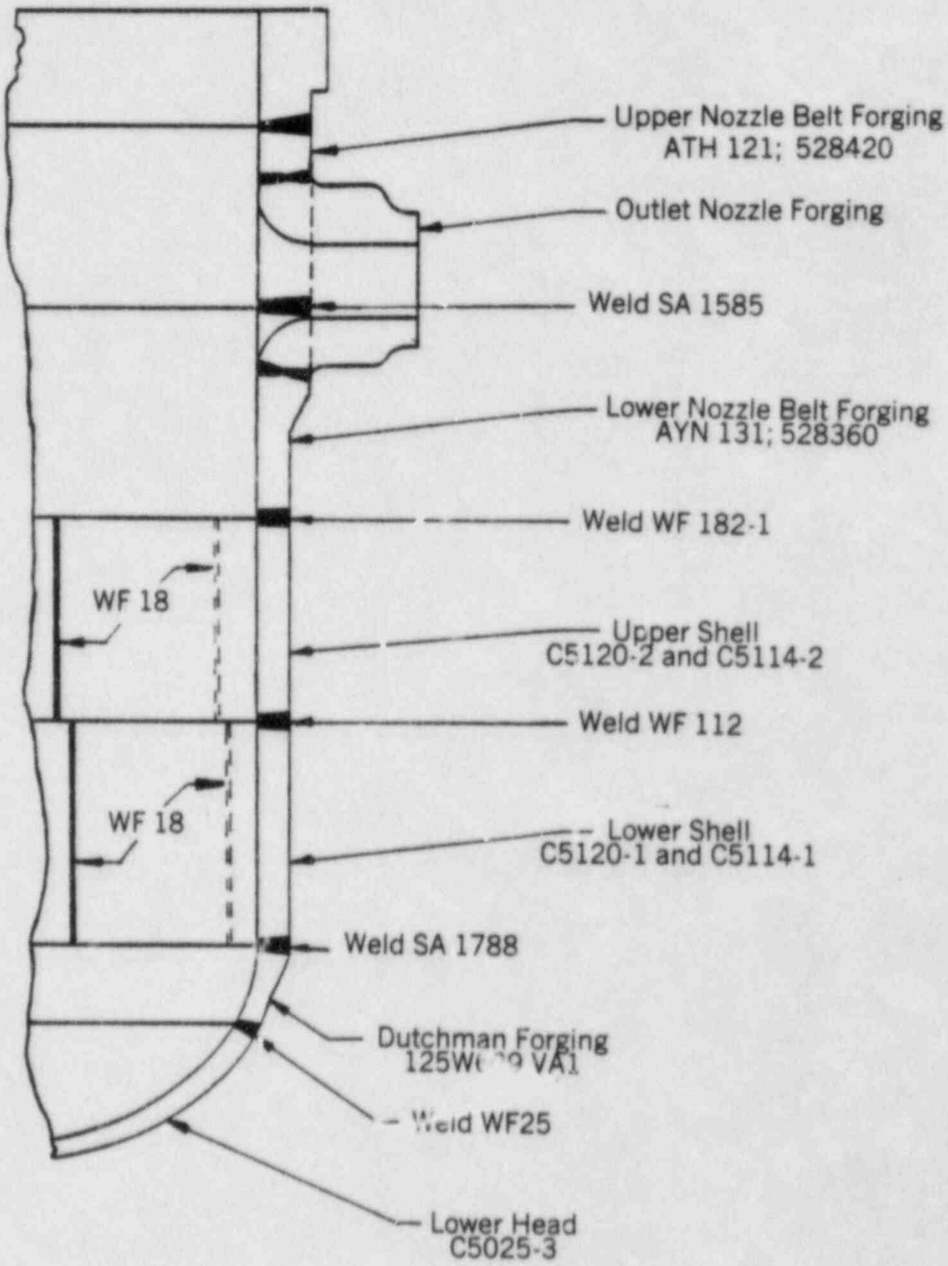


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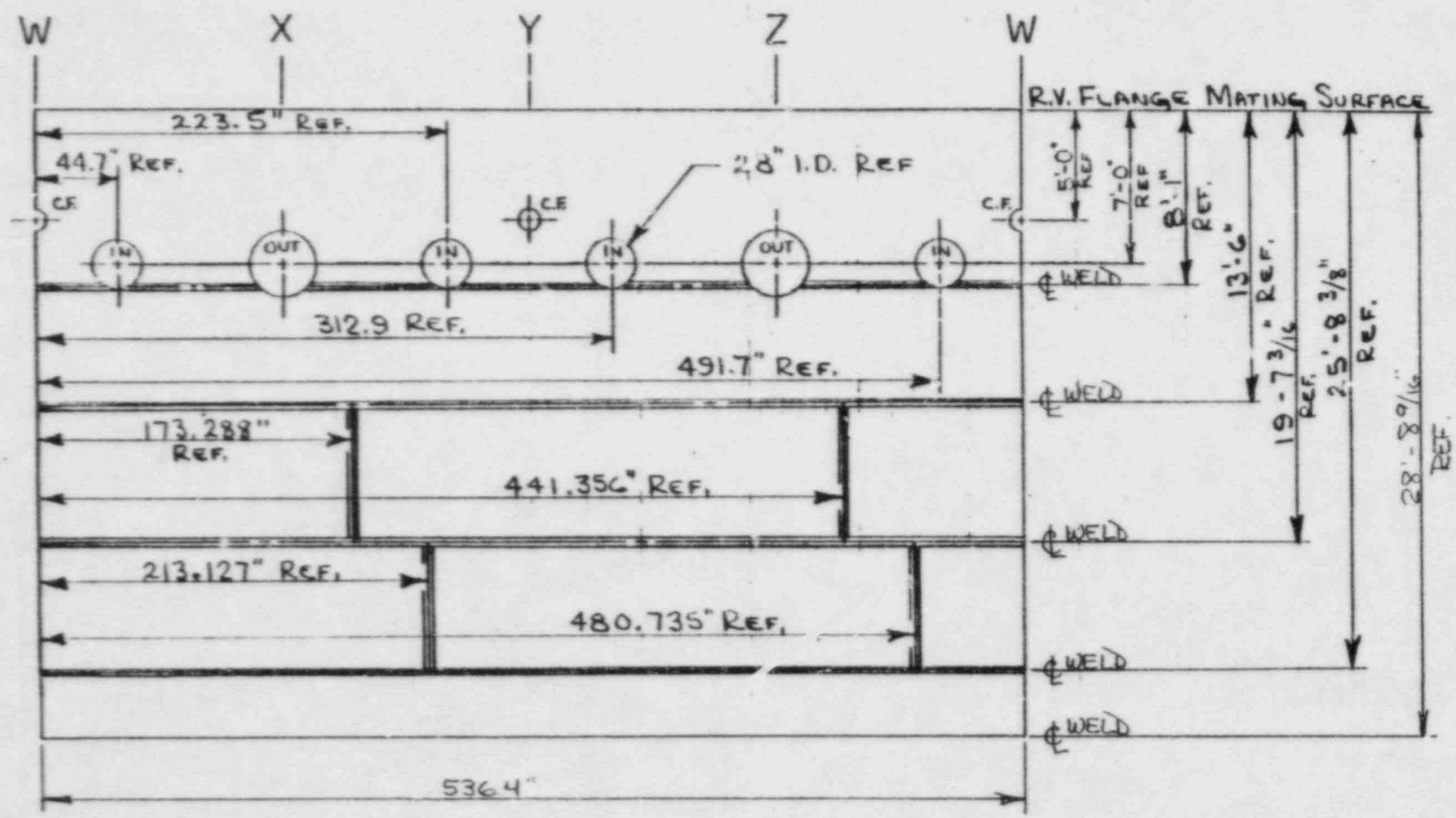
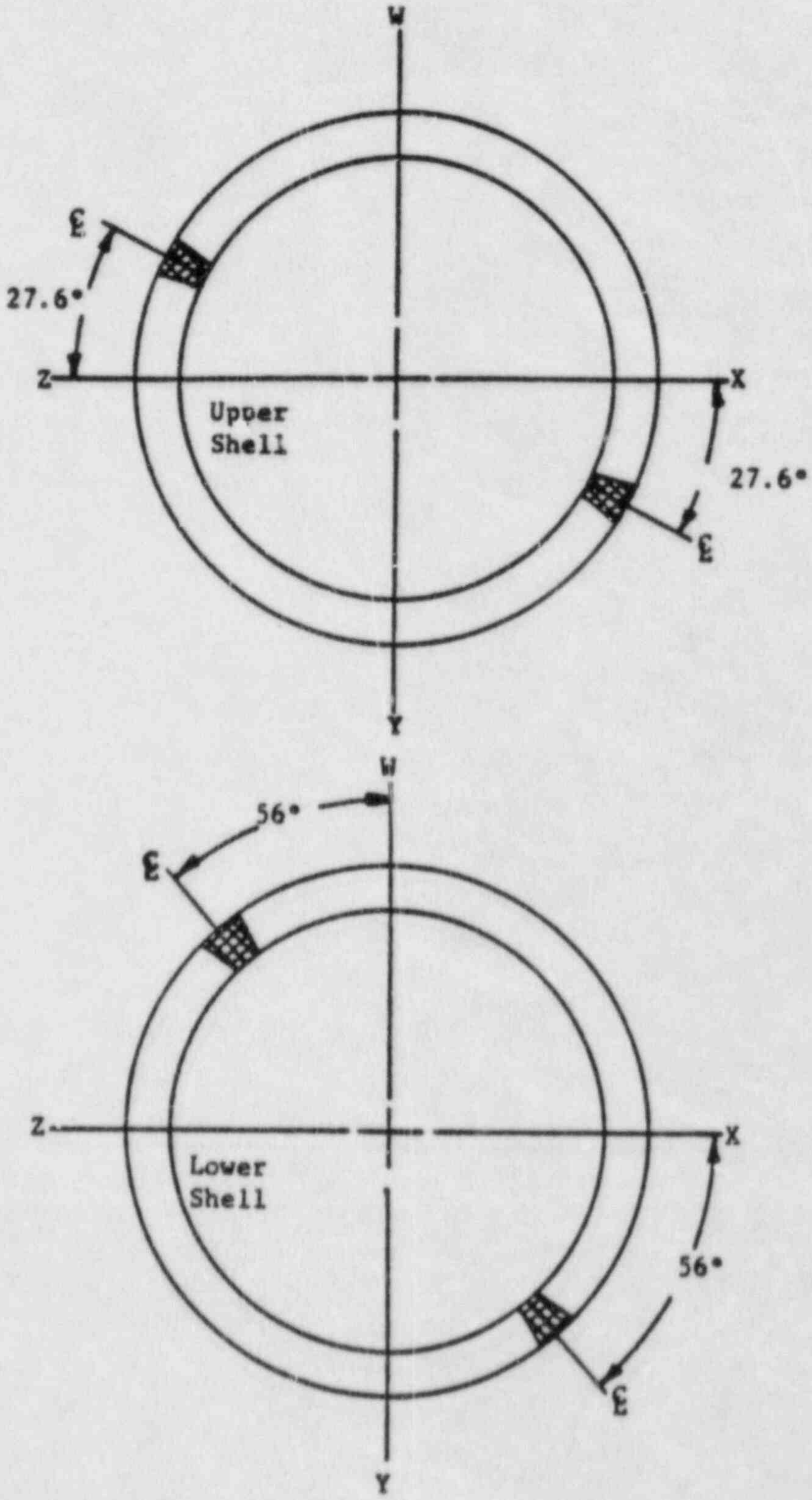
Figure 3.6-2. Reactor Vessel Weldment Locations in Arkansas Nuclear One Unit 1



Dimensions are based on inside travel

Figure 3.6-3. The Location of Longitudinal Welds in the Upper and Lower Shell Courses in Arkansas Nuclear One Unit 1



3.7. Oconee Unit 3 (NSS-9)

Table 3.7-1. General Plant Information for Oconee Unit 3 (NSS-9)

Plant Designation: Oconee Nuclear Station, Unit 3

Plant Location: Seneca, South Carolina

Start-up Date (on power grid): 9/11/74

Owner Utility: Duke Power Company

B&W Contract Number: 620-0009

Reactor Vessel Information

ASME Code Date: 1965 edition including Addenda through Summer 1967

Fabrication Site: Barberton, Ohio and Mt. Vernon, Indiana

Fabrication Schedule:

Start Date -- 6/7/68
Ship Date -- 7/16/71

Vessel Materials:

Nozzle Belt Forgings -- A 508 Cl. 2 modified by ASME Code Case 1332-3
Beltline Base Materials -- A 508 Cl. 2 modified by ASME Code Case 1332-4
Beltline Weld Materials -- ASA weld; MnMoNi wire, Linde 80 flux

Location of Materials in Vessel: See Figure 3.7-1.

Chemical Composition of Materials: See Table 3.7-2.

Heat Treatment of Materials: See Table 3.7-3.

Vessel Stress Relief: 1100-1150F for 29-3/4 h

Table 3.7-1. Chemical Composition of Beltline Region Materials for Oconee Unit 3

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
	Chemical Composition, wt %									
4680	0.21#	0.67	0.009	0.012	0.22	0.91	0.37	0.56	--	0.01 V, 0.03 Co Check analysis
	0.21#	0.67	0.009	0.012	0.21	0.91	0.36	0.57	--	0.02 V, 0.03 Co Check analysis
	0.21	0.62	0.012	0.009	0.26	0.84	0.36	0.60	--	0.02 V, 0.019 Co Ladle analysis
AWS 192; 522314	0.21*	0.58	0.011	0.015	0.24	0.73	0.30	0.60	0.01	<0.01 V, 0.01 Co Product analysis
	0.22	0.65	0.008	0.011	0.22	0.73	0.30	0.56	0.04	0.007 V, 0.01 Co Ladle analysis
	0.22	0.64	0.009	0.014	0.21	0.73	0.32	0.57	0.04	0.007 V, 0.01 Co Ladle analysis
ANK 191; 522194	0.24*	0.72	0.014	0.012	0.21	0.76	0.34	0.62	0.02	<0.01 V, 0.01 Co Product analysis
	0.27	0.77	0.013	0.010	0.23	0.75	0.34	0.57	0.04	<0.01 V, 0.01 Co Ladle analysis
	0.27	0.76	0.012	0.010	0.25	0.75	0.34	0.57	0.04	<0.01 V, 0.01 Co Ladle analysis
417543-1	0.20*	0.66	0.011	0.012	0.27	0.85	0.43	0.59	--	0.019 Co Product analysis
	0.20	0.67	0.011	0.012	0.27	0.85	0.42	0.59	--	0.01 V, 0.019 Co Ladle analysis
WF 200	0.07*	1.60	0.010	0.015	0.48	0.63	0.14	0.40	0.24	BAW-1500, Table 10

Table 3.7-2. (Cont'd)

	Chemical Composition, wt %									
Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
WF 200 (Cont'd)	0.08	1.69	0.014	0.013	0.45	0.63	0.14	0.40	0.24	BAW-1500, Table 6
	0.069	1.60	0.010	0.015	0.477	0.64	--	0.59	0.26	Weld qualification
WF 67	0.08*	1.55	0.021	0.016	0.58	0.60	0.09	0.39	0.24	BAW-1500, Table 10
	0.08	1.55	0.018	0.018	0.52	0.60	0.09	0.39	0.24	BAW-1500, Table 6
	0.064	1.49	0.014	0.017	0.54	0.57	0.02	0.41	0.27	Weld qualification
WF 70	0.09*	1.63	0.018	0.009	0.54	0.59	0.10	0.40	0.35	BAW-1500, Table 10
	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	BAW-1500, Table 6
	0.07	1.60	0.014	0.011	0.48	0.46	--	0.40	0.27	Weld qualification
WF 169-1	0.08*	1.56	0.016	0.016	0.45	0.63	0.08	0.37	0.18	BAW-1500, Table 10
	0.075	1.58	0.014	0.013	0.45	0.59	0.02	0.42	0.106	Weld qualification

*Analysis to be used for licensing.
#Average of multiple analysis to be used for licensing.

Table 3.7-3. Heat Treatment of Beltline Region Materials for Oconee Unit 3

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
4680 Lower nozzle belt forging	4680	NA	A 508-64 Cl. 2; mod. to ASME Code Case 1332-3	Rotterdam	1685F for 7 h, AC; 1220F for 16 h, FC; 1675F for 7 h, WQ; 1220F for 15 h, AC.
AWS 192 Upper shell	522314	NA	A 508-64 Cl. 2; mod. to ASME Code Case 1332-4	Ladish	1640F ±20F for 40 h, WQ; 1590F ±20F for 4 h, WQ; 1240F ±20F for 10 h, WQ.
ANK 191 Lower shell	522194	NA	A 508-64 Cl. 2; mod. to ASME Code Case 1332-4	Ladish	1640F ±20F for 4 h, WQ; 1590F ± 20F for 4 h, WQ; 1250F ±20F for 10 h, WQ.
417543-1 Dutchman forging	417543-1	NA	A 508 Cl. 2	Klockner-Werke AG	Unknown.
WF 200 NB to US circ. weld	821T44	Linde 80 8773		Union Carbide	
WF 67 US to LS circ. weld	72442	Linde 80 8669		Amer. Chain & Cable	

Table 3.7-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
WF 70 US to LS circ. weld	72105	Linde 80 8669		Amer. Chain & Cable	
WF 169-1 LS to Dutchman circ. weld	8T1554	Linde 80 8754		US Steel	

(a)NA denotes not applicable.

(b)FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Note: Welds and base metal may have received intermediate stress reliefs. All received full vessel stress relief of 1100-1150F for 29-3/4 h.

Table 3.7-4. Unirradiated Mechanical Properties of Beltline Region Materials for Oconee Unit 3

Material ID	Heat treatment	Toughness properties							Tensile properties					Notes
		Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Orientation	YS, ksi	UTS, ksi	El, %	RA, %	
4680	As per Table		117							78.2	95.2	26.6	68.5	Supplier Test Report data
	3.7-3, plus		111							78.2	95.3	27.0	67.6	
	1125F for 60		109											
	h, FC		113											
			49											
			101											
AWS 192	As per Table			-55	-30	90	+40	+40		63.4	86.6	26.0	68.0	Supplier Test Report data
	3.7-3, plus			-50	-25	100	+30	+30		61.8	84.8	26.0	70.0	-- three areas tested
	1125F ±25F for			-40	-18	97	+30	+30		60.8	84.8	27.0	70.0	
	25 h, FC													
ANK 191	As per Table			-10	+10	110	+20	+20		64.6	89.3	26.0	27.0	Supplier Test Report data
	3.7-3, plus			-2	+20	125	+40	+40		62.5	89.3	27.0	70.0	-- three areas tested
	1125F ±25F for									62.0	87.0	26.0	27.0	
	25 h, FC			+5	+25	123	+20	+20		61.3	86.5	26.0	70.0	
417543-1	Unknown, ex-		98							70.8	89.9	29.1	69.6	Supplier Test Report data
	cept for 1100-		109							69.6	89.9	29.5	69.6	
	1150F for 60		79											
	h, FC		85											
			108											
			67											
WF 200	1100-1150F for		36								84.5			Weld Qualification Test
	48 h, FC		35											data
			26											
WF 67	1100-1150F for		29							64.0	81.5	31.3	65.8	Weld Qualification Test
	48 h, FC		35											data
			30											
WF 70	1100-1150F for		39							69.0	85.5	25.8	64.7	Weld Qualification Test
	48 h, FC		35											data
			44											
WF 169-1	1100-1150F for		42							66.9	82.5	26.6	66.0	Weld Qualification Test
	8 6 h cycles,		29											data
	FC		46											

Figure 3.7-1. The Location and Identification of Materials Used in the Fabrication of the Reactor Pressure Vessel for Oconee Unit 3

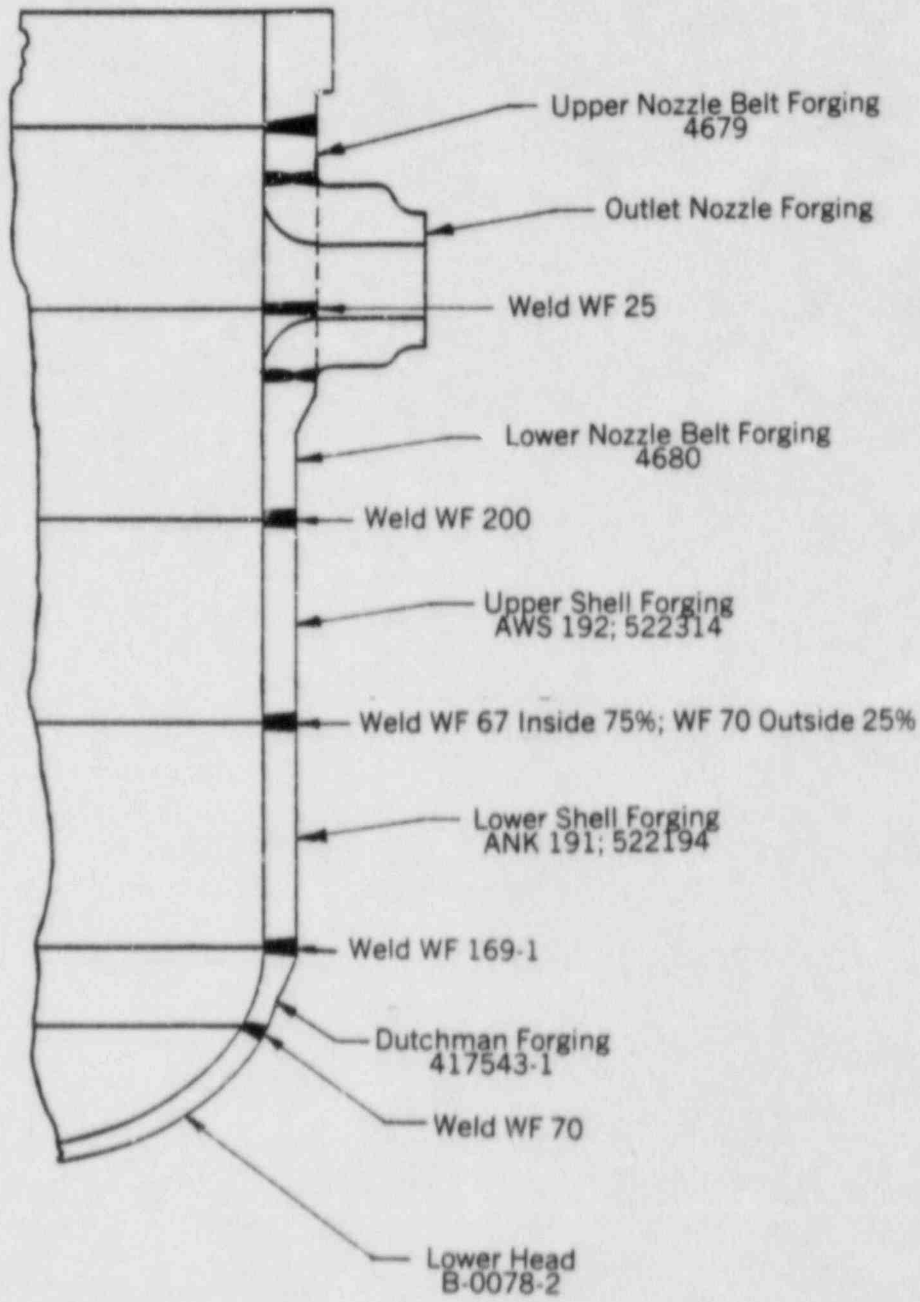
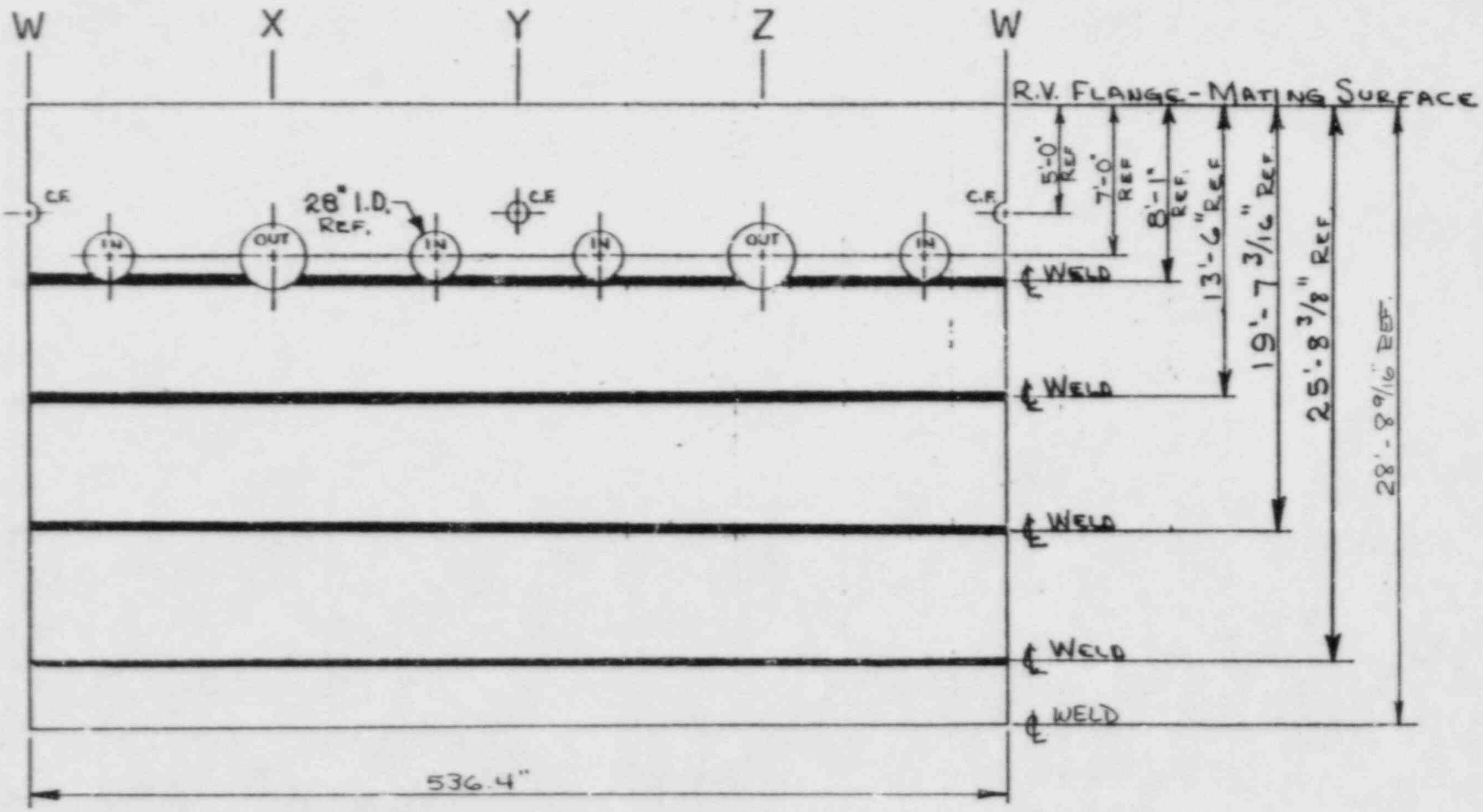


Figure 3.7-2. Reactor Vessel Weldment Locations in Oconee Unit 3



3.8. Rancho Seco Unit 1 (NSS-11)

Table 3.8-1. General Plant Information for Rancho Seco Unit 1 (NSS-11)

Plant Designation: Rancho Seco Unit 1

Plant Location: Clay Station, California

Start-up Date (on power grid): 10/13/74

Owner Utility: Sacramento Municipal Utility District

B&W Contract Number: 620-0011

Reactor Vessel Information

ASME Code Date: 1965 edition including Addenda through Summer 1967

Fabrication Site: Barberton, Ohio and Mt. Vernon, Indiana

Fabrication Schedule:

Start Date -- 4/9/69
Ship Date -- 10/21/71

Vessel Materials:

Nozzle Belt Forgings -- A 508 Cl. 2 modified by ASME Code Case 1332-3
Beltline Base Materials -- SA-533 Gr. B, Cl. 1
Beltline Weld Materials -- ASA weld; MnMoNi wire, Linde 80 flux

Location of Materials in Vessel: See Figure 3.8-1.

Chemical Composition of Materials: See Table 3.8-2.

Heat Treatment of Materials: See Table 3.8-3.

Vessel Stress Relief: Upper Shell -- 1100-1150F for 27 h (cumulative)
Lower Shell -- 1100-1150F for 27-3/4 h (cumulative)

Table 3.8-2. Chemical Composition of Beltline Region Materials for Rancho Seco Unit 1

	Chemical Composition, wt %									
Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
FV 4823; BV 3062+ZV 4281	0.18	0.70	0.009	0.005	0.25	0.68	0.44	0.60	0.15	0.01 V, 0.008 Co ZV 4281 analysis
	0.21	0.73	0.010	0.007	0.25	0.63	0.42	0.61	0.15	0.01 V, 0.005 Co BV 3062 analysis
	0.19#	0.74	0.009	0.006	0.23	0.66	0.39	0.60	0.15	0.01 V, 0.008 Co Forging check analysis
	0.19#	0.75	0.009	0.006	0.23	0.68	0.40	0.62	0.15	0.01 V, 0.008 Co Forging check analysis
C5062-1	0.20	1.26	0.013	0.017	0.15	0.60	0.14	0.55	0.12	0.007 Co
C5062-2	0.20	1.26	0.013	0.017	0.15	0.60	0.14	0.55	0.12	0.007 Co
C5070-1	0.20	1.33	0.010	0.015	0.19	0.58	0.20	0.52	0.10	0.019 Co
C5070-2	0.20	1.33	0.010	0.015	0.19	0.58	0.20	0.52	0.10	0.019 Co
123X264VA1	0.22	0.66	0.009	0.009	0.24	0.70	0.34	0.60	0.06	0.01 V, 0.004 Co Ladle analysis
	0.20#	0.65	0.01	0.008	0.22	0.69	0.33	0.59	--	0.02 V, 0.009 Co Forging check analysis
	0.20#	0.66	0.01	0.007	0.21	0.69	0.32	0.59	--	0.02 V, 0.007 Co Forging check analysis

Table 3.8-2. (Cont'd)

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
	Chemical Composition, wt %									
WF 233	0.05*	1.45	0.021	0.015	0.42	0.68	0.08	0.44	0.29	BAW-1500, Table 10
	0.053	1.60	0.015	0.016	0.44	0.55	--	0.47	0.22	Weld qualification
WF 154	0.07*	1.54	0.013	0.016	0.42	0.59	0.07	0.40	0.31	BAW-1500, Table 10
	0.08	1.48	0.016	0.016	0.53	0.59	0.06	0.40	0.31	BAW-1500, Table 6
	0.072	1.50	0.015	0.021	0.45	0.59	--	0.30	0.20	Weld qualification
WF 29	0.08*	1.65	0.015	0.012	0.42	0.63	0.05	0.38	0.23	BAW-1500, Table 10
	0.079	1.55	0.017	0.010	0.42	0.27	--	0.33	0.16	Weld qualification
WF 70	0.09*	1.63	0.018	0.009	0.54	0.59	0.10	0.40	0.35	BAW-1500, Table 10
	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	BAW-1500, Table 6
	0.07	1.60	0.014	0.011	0.48	0.46	--	0.40	0.27	Weld qualification

*Analysis to be used for licensing.
#Average of multiple analysis to be used for licensing.

Table 3.8-3. Heat Treatment of Beltline Region Materials for Rancho Seco Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
FV 4823 Lower nozzle belt forging	BV 3062+ ZV 4281	NA	A 508 Cl. 2 mod. by ASME Code Case 1332-3	Midvale-Heppenstall	1750F for 12 h, AC; 1250F for 18 h, FC; 1625F for 11 h, WQ; 1215F for 22 h, AC
C5062-1 Upper shell	C5062-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ
C5062-2 Upper shell	C5062-2	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ
C5070-1 Lower shell	C5070-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ
C5070-2 Lower shell	C5070-2	NA	SA-533 Gr. B Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ.
123X264VA1 Dutchman forging	123X264	NA	A 508 Cl. 2 mod. by ASME Code Case 1332-3	Bethlehem Steel	1550F for 22 h, WQ; 1230F for 14 h, AC
WF 233 NB to US circ. weld	T29744	Linde 80 8790		Union Carbide Linde	

Table 3.8-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
WF 154 US to LS circ. weld	406L44	Linde 80 8720		Union Carbide	
WF 29 US and LS long. welds	72102	Linde 80 8650		Amer. Chain & Cable	
WF 70 LS long. weld	72105	Linde 80 8669		Amer. Chain & Cable	

(a) NA denotes not applicable.

(b) FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Note: Welds and base metal may have received intermediate stress reliefs. All received full vessel stress relief. Upper shell cumulative received 27 h at 1100-1150F. Lower shell cumulative received 27-3/4 h at 1100-1150F. US to LS circ. weld stress relief is believed to be 28 h based on RVSP weld records.

Table 3.8-4. Unirradiated Mechanical Properties of Beltline Region Material for Rancho Seco Unit 1

Material ID	Heat treatment	Toughness properties							Tensile properties					Notes
		Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Orientation	YS, ksi	UTS, ksi	El, %	RA, %	
FV 4823	As per Table 3.8-3, plus 1125F for 60 h, FC		108 102 114 145 118 124							74.5 62.0	94.0 83.5	30.0 26.2	74.5 71.5	Supplier Test Report data
	As per Table 3.8-3, plus unknown stress relief	LT TL		<+70 <+70	<+70 <+70	15[illegible] 130	+10	+10						Supplementary Mt. Vernon tests of excess shell course material
C5062-1	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		55 30 65							62.6	83.5	24.0		Supplier Test Report data
C5062-1	As per Table 3.8-3, plus 1100-1150F for 40 h, FC	1/4T location		-22	+8	130	-10	-10	1/4T location	64.5 64.0	85.0 85.0	28.9 28.1	71.5 70.0	Mt. Vernon Qualification Test data
C5062-1	As per Table 3.8-3, plus 1100-1150F for 28 h, FC	TL		+15	+64	90	-10	+4						Supplementary Mt. Vernon tests of excess surveillance program material
C5062-2	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		60 73 48							67.6	88.8	23.0		Supplier Test Report data
C5062-2	As per Table 3.8-3, plus 1100-1150F for 40 h, FC	1/4T location		-16	+6	140	-10	-10	1/4T location	63.5 65.0	84.0 85.5	29.7 27.3	71.1 70.9	Mt. Vernon Qualification Test data

Table 3.8-4. (Cont'd)

		Toughness properties							Tensile properties					
Material ID	Heat treatment	Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
C5062-2	As per Table 3.8-3, plus unknown stress relief	TL		<+30	+48	87								Supplementary Mt. Vernon tests of excess shell course material
C5070-1	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		46 56 61							69.3	92.5	26.0		Supplier Test Report data
C5070-1	As per Table 3.8-3, plus 1100-1150F for 40 h, FC	1/4T location		-4	+10	133	-20	-20	1/4T location	65.8 67.5	86.2 89.2	27.3 26.6	61.2 69.4	Mt. Vernon Qualification Test data
C5070-1	As per Table 3.8-3, plus 1100-1150F for 28 h, FC	TL		+15	+60	92	-20	0						Supplementary Mt. Vernon tests of excess surveillance program material
C5070-2	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC; 1100-1150F for 60 h, FC		75 60 63							72.1	89.4	26.0		Supplier Test Report data
C5070-2	As per Table 3.8-3, plus 1100-1150F for 40 h, FC	1/4T location		-18	+16	134	-20	-20	1/4T location	69.0 70.8	89.2 91.0	28.1 26.6	69.0 70.0	Mt. Vernon Qualification Test data
C5070-2	As per Table 3.8-3, plus unknown stress relief	TL		<+30	+50	[illegible]								Supplementary Mt. Vernon tests of excess shell course material

Table 3.8-4. (Cont'd)

Material ID	Heat treatment	Toughness properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
123X264VA1	As per Table 3.8-3, plus 1125F for 60 h, FC		82							72.0	94.5	25.0	69.2	Supplier Test Report data
			49							72.0	95.5	25.0	69.7	
			86											
			88											
			85											
			78											
WF 233	1100-1150F for 48 h, FC		43							63.8	80.5			Weld Qualification Test data
			30											
			26											
WF 154	1100-1150F for 48 h, FC		41							65.2	81.5	26.6	66.0	Weld Qualification Test data
			37											
			43											
WF 29	1100-1150F for 48 h, FC		49							67.8	83.0	26.6		Weld Qualification Test data
			39											
			45											
WF 70	1100-1150F for 48 h, FC		39							69.0	85.5	25.8	64.7	Weld Qualification Test data
			35											
			44											

Figure 3.8-1. The Location and Identification of Materials Used in the Fabrication of the Reactor Pressure Vessel for Rancho Seco Unit 1

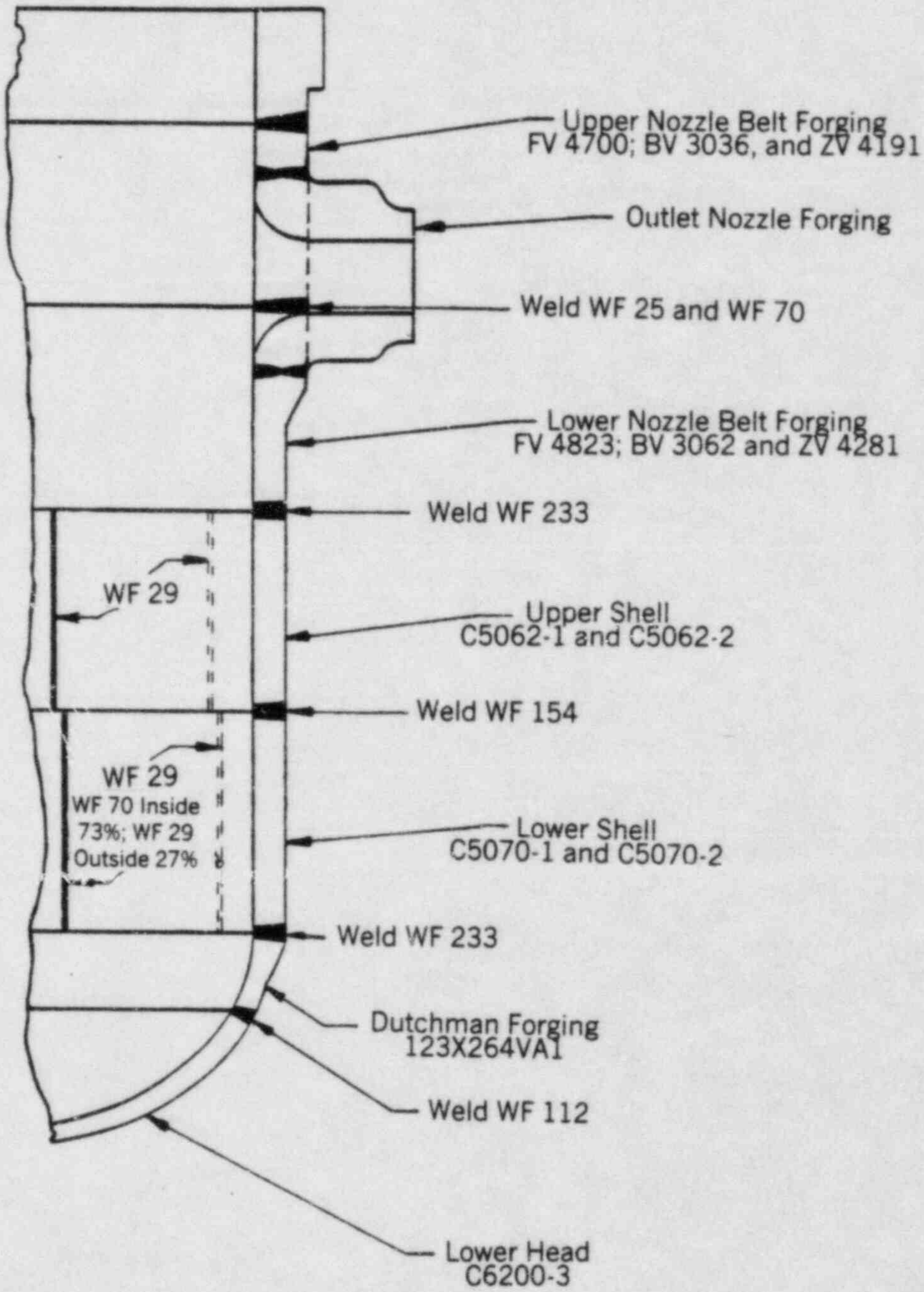
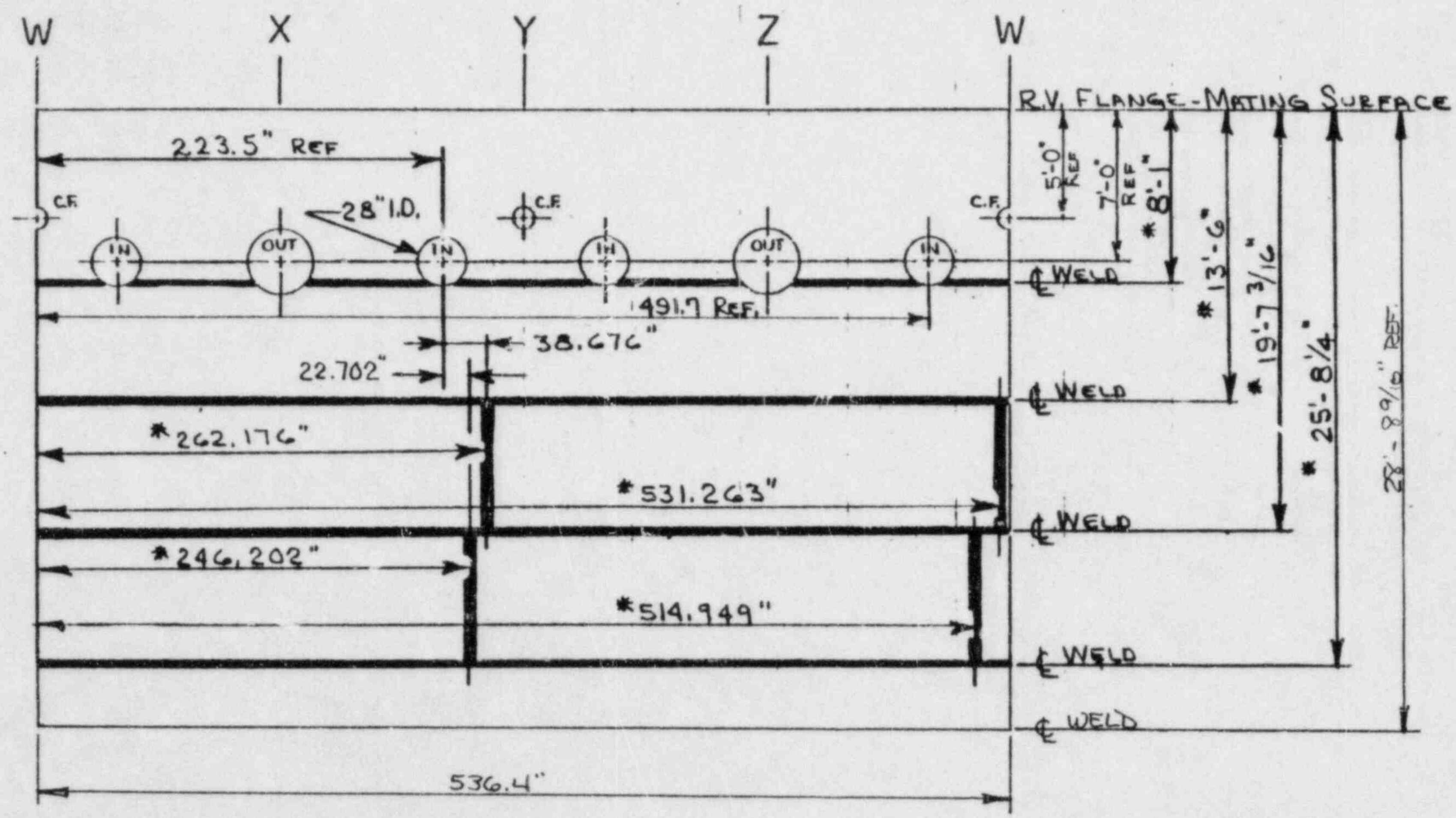


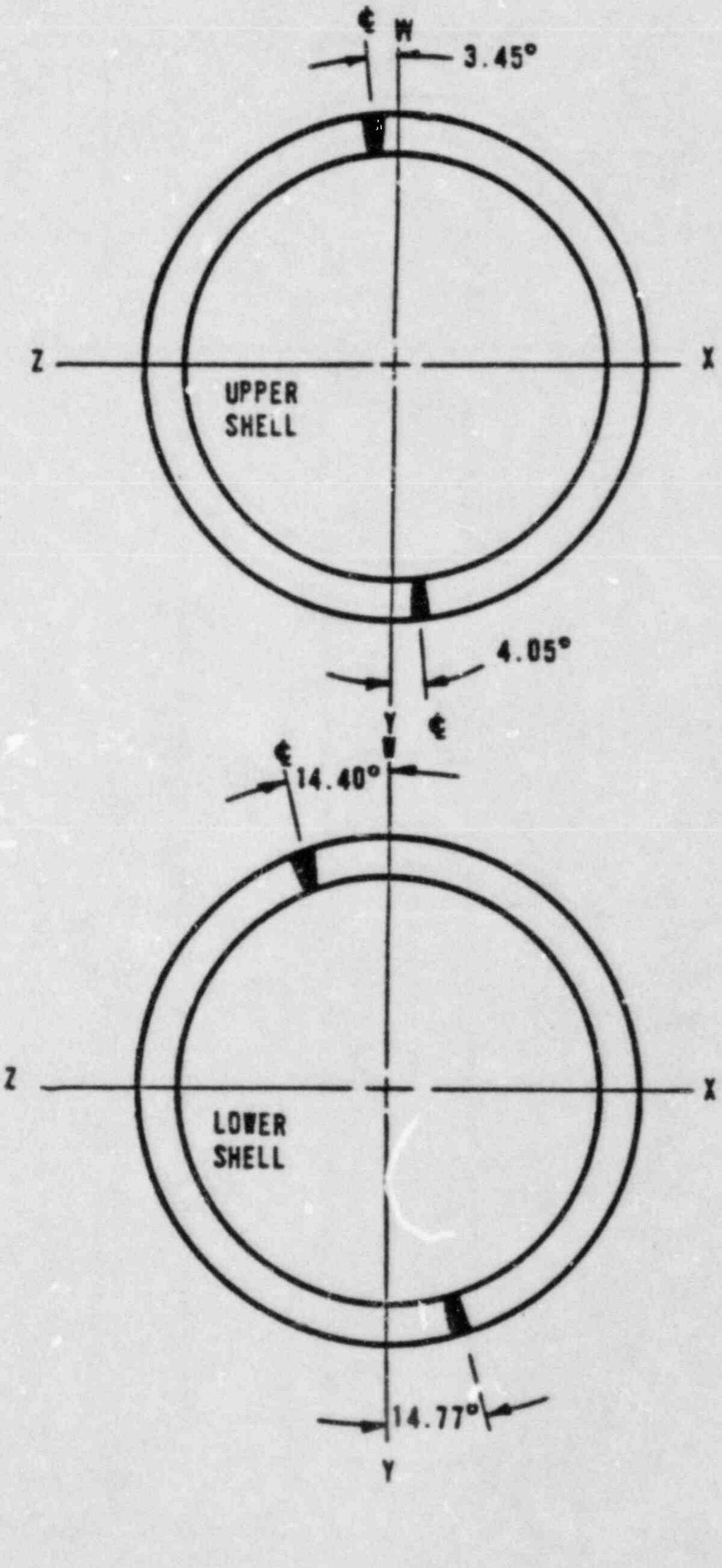
Figure 3.8-2. Reactor Vessel Weldment Locations in Rancho Seco Unit 1



Dimensions are based on inside travel

*From as-built drawing

Figure 3.8-3. The Location of Longitudinal Welds in the Upper and Lower Shell Courses in Rancho Seco Unit 1



3.9. Davis-Besse Unit 1 (NSS-14)

Table 3.2-1. General Plant Information for Davis-Besse Unit 1 (NSS-14)

Plant Designation: Davis-Besse Unit 1

Plant Location: Oak Harbor, Ohio

Start-up Date (on power grid): 8/28/77

Owner Utility: Toledo Edison Company

B&W Contract Number: 620-0014

Reactor Vessel Information

ASME Code Date: 1968 edition including Addenda through Summer 1968

Fabrication Site: Barberton, Ohio and Mt. Vernon, Indiana

Fabrication Schedule:

Start Date -- 7/7/70
Ship Date -- 11/11/72

Vessel Materials:

Nozzle Belt Forgings -- A 508 Cl. 2 modified by ASME Code Case 1332-4
Beltline Base Materials -- A 508 Cl. 2 modified by ASME Code Case 1332-4
Beltline Weld Materials -- ASA weld; MnMoNi wire, Linde 80 flux

Location of Materials in Vessel: See Figure 3.9-1.

Chemical Composition of Materials: See Table 3.9-2.

Heat Treatment of Materials: See Table 3.9-3.

Vessel Stress Relief: 1100-1150F for 15 h

Table 3.9-2. Chemical Composition of Beltline Region Materials for Davis-Besse Unit 1

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
ADB 203; 123Y317	0.23	0.64	0.010	0.011	0.24	0.70	0.36	0.56	--	0.04 V, 0.010 Co Ladle analysis
	0.23*	0.70	0.007	0.009	0.29	0.68	0.39	0.63	0.04	0.05 V, 0.01 Co Product analysis
AKJ 233; 123X244	0.24	0.73	0.007	0.006	0.28	0.73	0.37	0.61	--	0.04 V, 0.006 Co Ladle analysis
	0.26*	0.68	0.004	0.006	0.30	0.77	0.38	0.64	0.04	0.04 V, 0.01 Co Product analysis
BCC 241; 5P4086	0.22	0.64	0.012	0.016	0.28	0.83	0.34	0.59	0.04	0.03 V, 0.01 Co Ladle analysis
	0.21	0.63	0.007	0.010	0.25	0.76	0.35	0.62	0.04	0.03 V, 0.01 Co Ladle analysis
	0.22*	0.63	0.011	0.011	0.27	0.81	0.32	0.63	0.02	0.02 V, 0.01 Co Product analysis
122Y384VA1	0.21	0.61	0.008	0.008	0.25	0.72	0.32	0.61	0.08	0.01 V, 0.007 Co Ladle analysis
	0.20#	0.65	0.01	0.007	0.24	0.75	0.39	0.57	--	0.02 V, 0.010 Co Check analysis
	0.19#	0.65	0.01	0.007	0.24	0.74	0.40	0.57	--	0.02 V, 0.011 Co Check analysis
WF 232	0.055	1.45	0.011	0.007	0.51	0.69	--	0.30	0.138	Weld qualification

Table 3.9-2. (Cont'd)

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
WF 233	0.05*	1.45	0.021	0.015	0.42	0.68	0.08	0.44	0.29	BAW-1500, Table 10
	0.053	1.60	0.015	0.016	0.44	0.55	--	0.47	0.22	Weld qualification
WF 182-1	0.08#	1.69	0.014	0.013	0.45	0.63	0.14	0.40	0.24	BAW-1500, Table 10
	0.08	1.69	0.014	0.013	0.45	0.63	0.14	0.40	0.24	BAW-1500, Table 6
	0.071	1.60	0.014	0.015	0.30	0.59	--	0.42	0.18	Weld qualification

*Analysis to be used for licensing.
#Average of multiple analysis to be used for licensing.

Table 3.9-3. Heat Treatment of Beltline Region Materials for Davis-Besse Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
ADB 203 Nozzle belt forging	123Y317	NA	A 508-64 Cl. 2, mod. to ASME Code Case 1332-4	Ladish	1640F ±10F for 6 h, WQ; 1590F ±10F for 6 h, WQ; 1240F ±10F for 14 h, WQ
AKJ 233 Upper shell	123X244	NA	A 508-64 Cl. 2, mod. to ASME Code Case 1332-4	Ladish	1640F ±10F for 4 h, WQ; 1590F ±10F for 4 h, WQ; 1240F ±10F for 6 h, AC
BCC 241 Lower shell	5P4086	NA	A 508-64 Cl. 2, mod. to ASME Code Case 1332-4	Ladish	1640F ±10F for 4 h, WQ; 1590F ±10F for 4 h, WQ; 1240F ±10F for 5 h, AC
122Y384VA1 Dutchman forging	122Y384	NA	A 508-64 Cl. 2, mod. to ASME Code Case 1332-4	Bethlehem Steel	1575F for 22 h, WQ; 1220F for 33 h, AC
WF 232 NB to US circ. weld and LS to Dutchman circ. weld	8T3914	Linde 80 8790		Raco	

Table 3.9-3. (Cont'd)

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
WF 233 NB to US circ. weld and LS to Dutchman circ. weld	T29744	Linde 80 8790		Union Carbide Linde	
WF 182-1 US to LS circ. weld	821T44	Linde 80 8754		Union Carbide Linde	

(a)NA denotes not applicable.

(b)FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Note: Welds and base metal may have received intermediate stress reliefs. All received full vessel stress relief of 15 h at 1100-1150F.

Table 3.9-4. Unirradiated Mechanical Properties of Beltline Region Materials for Davis-Besse Unit 1

Material ID	Heat treatment	Toughness properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
ADB 203	As per Table 3.9-3, plus 1100-1150F for 40 h, FC		71 70 67 118 113 102 +40F 102 111 76 130 123 133							73.6 74.6	93.9 95.9	25.0 25.0	68.0 71.0	Supplier Test Report data
ADB 203	As per Table 3.9-3, plus unknown stress relief	LT		+40	+55	132	+50	+50						Mt. Vernon Qualification Test data
ADB 203	As per Table 3.9-3, plus unknown stress relief	TL		+48	+65	134	+50	+50						Supplementary Mt. Vernon tests of excess shell course material
AKJ 233	As per Table 3.9-3, plus 1125 ±25F for 40 h, FC			-125 -40 -40	-100 -25 -10	165 185 160	0 +20 +20	0 +20 +20		76.6 75.8 71.5 76.4	94.6 94.3 89.8 96.4	24.0 24.0 26.0 23.0	73.0 70.0 72.0 71.0	Supplier Test Report data -- three areas tested
AKJ 233	As per Table 3.9-3, plus unknown stress relief	TL		-40	-6	144	+20	+20						Mt. Vernon Qualification Test data
AKJ 233	As per Table 3.9-3, plus 1100-1150F for 15-1/2 h, FC	TL		-15	+30	144	+20	+20						Supplementary Mt. Vernon tests of excess surveillance program material

Table 3.9-4. (Cont'd)

Material ID	Heat treatment	Toughness properties: Orientation	C_v +10F, ft-lbs	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile properties: Orientation	YS, ksi	UTS, ksi	El, %	RA, %	Notes
BCC 241	As per Table 3.9-3, plus 1125F ±25F for 40 h, FC			-25	0	130	+50	+50		77.4	96.4	23.0	68.0	Supplier Test Report data -- three areas tested
				-25	0	133	+40	+40		75.6	94.4	24.0	70.0	
										71.4	91.6	24.0	69.0	
				-30	-15	141	+40	+40		74.6	93.6	24.0	69.0	
BCC 241	As per Table 3.9-3, plus unknown stress relief	TL		-30	0	118	+50	+50						Mt. Vernon Qualification Test data
BCC 241	As per Table 3.9-3, plus 1100-1150F for 15-1/2 h, FC	TL		-14	+27	118	+50	+50						Supplementary Mt. Vernon tests of excess surveillance program material
122Y384VA1	As per Table 3.9-3, plus 1125F for 60 h, FC		118 111 104 136 113 118							66.2 65.0	89.5 87.0	25.5 27.0	72.7 72.4	Supplier Test Report data
WF 232	1100-1150F for 48 h, FC		25 31 35								81.2			Weld Qualification Test data
WF 233	1100-1150F for 48 h, FC		43 30 26							63.8	80.5			Weld Qualification Test data
WF 182-1	1100-1150F for 48 h, FC		36 33 44								71.0			Weld Qualification Test data
WF 182-1	Unknown stress relief			-28	+28	81	-20	-20						Mt. Vernon Qualification Test data
WF 182-1	1100-1150F for 15-1/2 h, FC	LT		+5	+62	81	-20	+2						Supplementary Mt. Vernon tests of excess surveillance program material

Figure 3.9-1. The Location and Identification of Materials Used in the Fabrication of the Reactor Pressure Vessel for Davis-Besse Unit 1

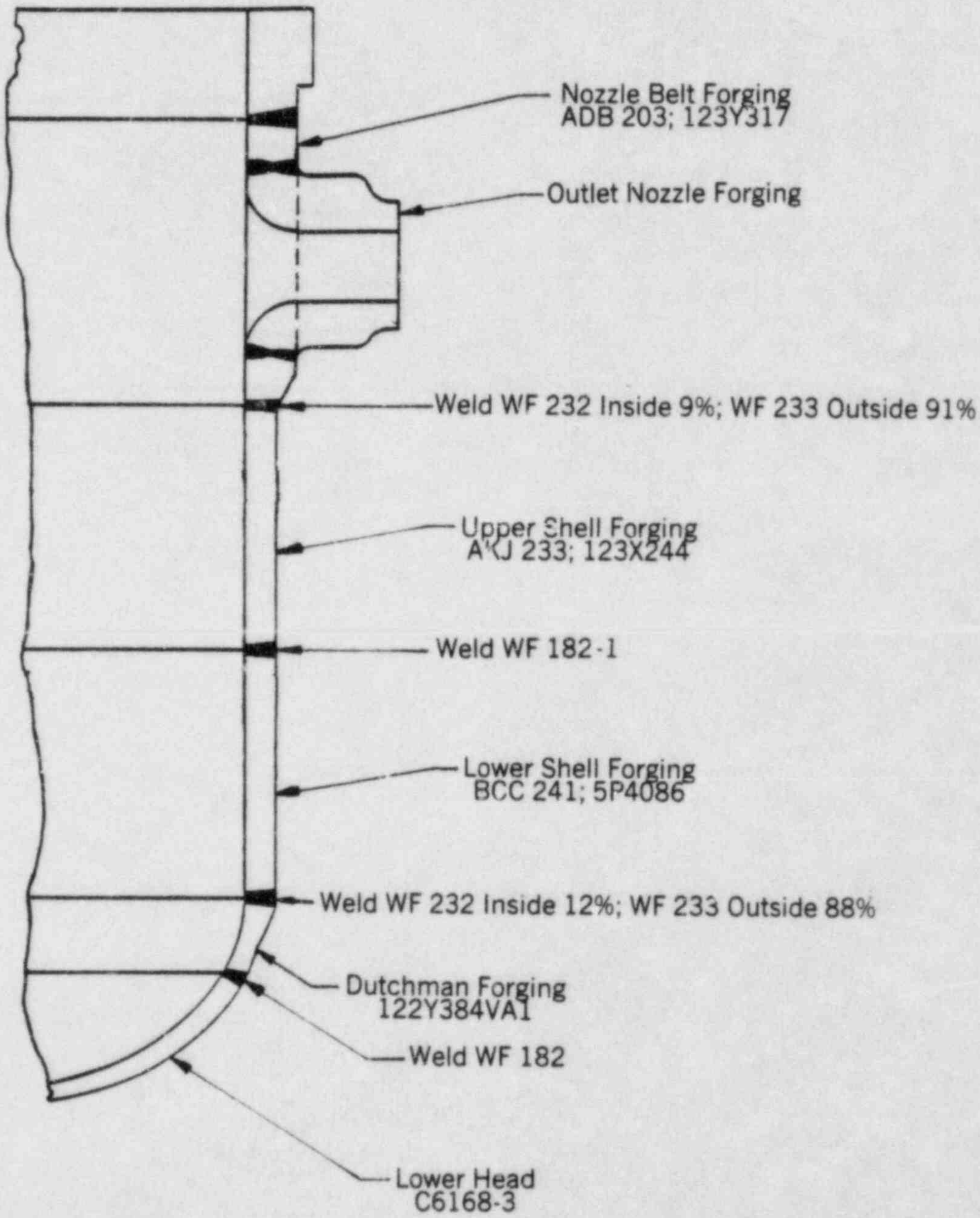
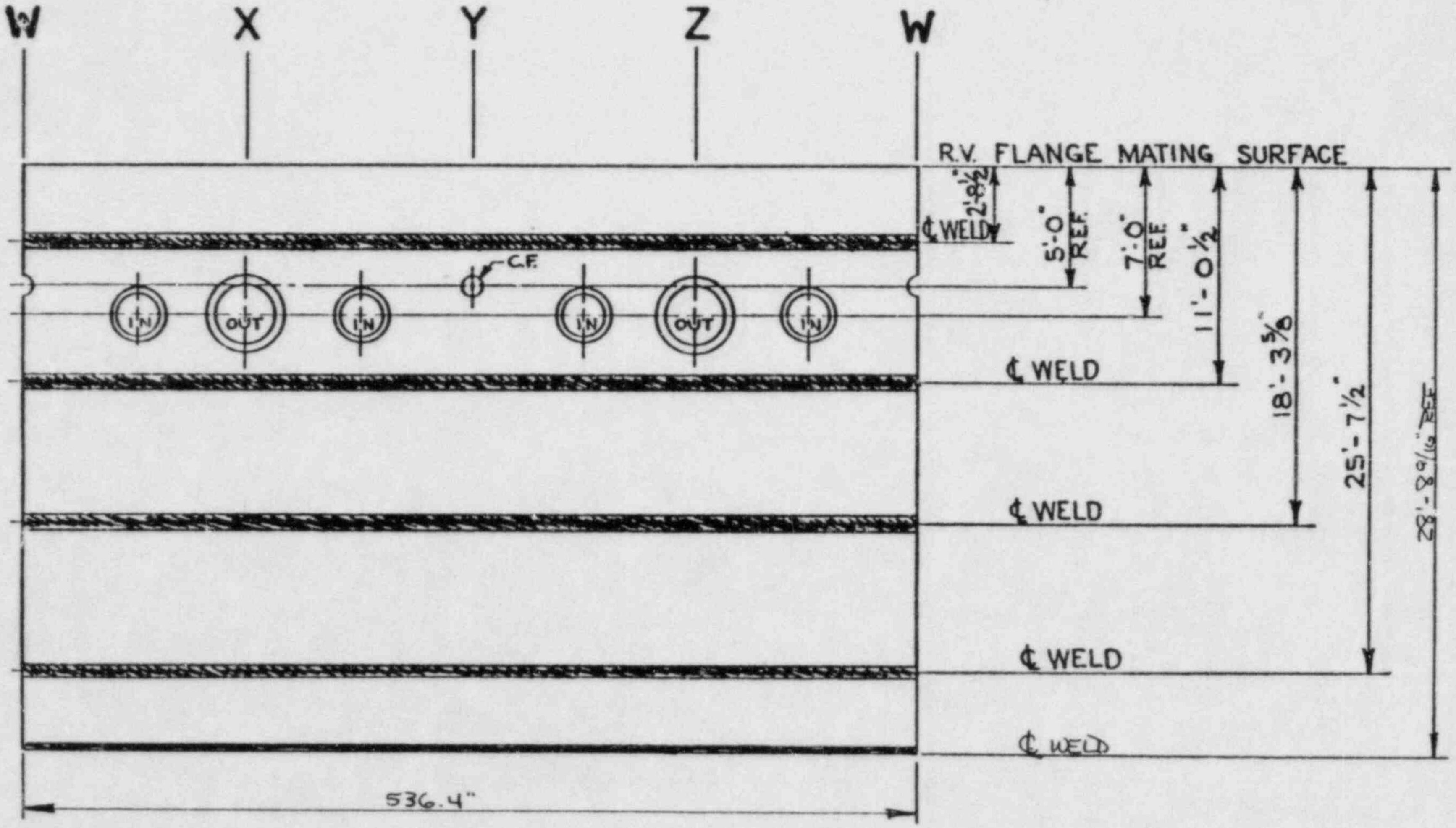


Figure 3.9-2. Reactor Vessel Weldment Locations in Davis-Besse Unit 1



4. PLANT-SPECIFIC RVSP INFORMATION

4.1. Description

This section contains information pertaining to the chemical compositions, heat treatments, specimen orientations, and unirradiated mechanical properties of the RVSP materials. Each subsection is devoted to the data pertaining to one RVSP, with information presented wholly in tabular format. Tables 4.n-1 present background information regarding documentation, surveillance program materials, and test specimens available in both unirradiated and irradiated conditions. Tables 4.n-2 list the chemical compositions for each RVSP material; as with Tables 3.n-2, the choice of licensing analysis should follow the convention developed in section 2.3.

Tables 4.n-3 list the heat treatments for the RVSP materials. Problems in tracing the documentation, similar to those for the reactor materials, were experienced here; the heat treatments (not including stress relief times) for base metals should be identical for both vessel and corresponding RVSP materials. Therefore, the only variables to be considered in a data analysis are the stress relief times of the vessel and RVSP materials. In the event that surveillance program heat treatments become a concern, Mt. Vernon fabrication records will be consulted to clarify matters. The information presented in Tables 4.n-3 is considered to be accurate.

Tables 4.n-4 list the unirradiated mechanical properties for the RVSP materials, based on tests of the baseline (unirradiated) specimens. These test data were originally reported in BAW capsule reports and are reproduced here for convenience. This information is considered to be a better representation of the mechanical properties of the vessel material than that given in section 3, since the heat treatments and/or stress relief times of the RVSP test specimens are closer to those of the vessel than are those of the earlier supplier and qualification test specimens. The orientations of the RVSP test specimens are known, and quality assurance documentation of the test data is available.

Tables 4.n-5 are presented to eliminate any confusion regarding test specimen orientation. The original test specimen sectioning diagrams were reviewed, and the orientation convention developed in section 2.4 is used. Previous data reports may contain inaccuracies; the orientations shown in this section are to be considered correct.

4.2. Oconee Unit 1 RVSP

Table 4.2-1. Plant-Specific RVSP Information for Oconee Unit 1

B&W Documentation: BAW-10006A, Rev. 3,[5] BAW-1543, Rev. 2[14], R&D Division Report 9000[6]

Number of Surveillance Capsules: Six

Irradiation Sites: Oconee Unit 1 and Crystal River Unit 3

Surveillance Program Materials:

- Base Metal -- C3265-1
  C2800-2
- HAZ Metal -- C3265-1
  C2800-2
- Weld Metal -- WF 112
- Correlation Metal -- HSST Plate 02

Surveillance Program Specimens:

Material	Orientation	Tensile	CVN
Baseline			
C3265-1	Longitudinal	6	21 (LT)
C3265-1	Transverse	6	21 (TL)
C2800-2	Longitudinal	6	19 (LT)
C2800-2	Transverse	6	20 (TL)
HAZ -- C3265-1	Longitudinal	6	0
HAZ -- C3265-1	Transverse	6	21 (TL); 20 (TS)
HAZ -- C2800-2	Longitudinal	6	0
HAZ -- C2800-2	Transverse	6	18 (TL); 23 (TS)
Weld -- WF 112		6	21 (TL)
Capsules A, C, E			
C3265-1	Longitudinal	4	8 (LT)
C3265-1	Transverse	0	4 (TL)
HAZ -- C3265-1	Transverse	0	8 (TL)
Weld -- WF 112		4	8 (TL)
HSST Plate 02	Longitudinal	0	8 (LT)

Table 4.2-1. (Cont'd)

Material	Orientation	Tensile	CVN
Capsules B, D, F			
C2800-2	Longitudinal	4	10 (LT)
C2800-2	Transverse	0	8 (TL)
HAZ -- C2800-2	Transverse	4	10 (TL)
HSST Plate 02	Longitudinal	0	8 (LT)

Table 4.2-2. Chemical Composition of Plant-Specific RVSP Materials for Oconee Unit 1

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
	Chemical Composition, wt %									
C3265-1	0.21	1.42	0.015	0.015	0.23	0.50	0.17	0.49	0.10	0.016 Co
C2800-2	0.20	1.40	0.012	0.017	0.20	0.63	0.13	0.50	0.11	0.014 Co
WF 112	0.08*	1.47	0.016	0.015	0.54	0.59	0.07	0.40	0.32	BAW-1500, Table 5
	0.08	1.47	0.016	0.015	0.54	0.59	0.07	0.40	0.31	BAW-1500, Table 10
	0.08	1.48	0.016	0.016	0.53	0.59	0.06	0.40	0.31	BAW-1500, Table 6
	0.08	1.50	0.024	0.006	0.60	0.58	--	0.51	0.22	Weld qualification
Correlation plate; HSST 02	0.23	1.39	0.013	0.013	0.21	0.64	--	0.50	0.17	ORNL-4463[15]

*Used for data analysis.

Table 4.2-3. Heat Treatment of Plant-Specific RVSP Materials for Oconee Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
C3265-1	C3265-1	NA	SA-302B, mod. to ASME Code Case 1339	Lukens	1625-1675F for 1 h/in., WQ; 1200-1250F for 1 h/in. AC. Probably also 1600-1650F for 9-3/4 h, BQ; 1200-1220F for 9-1/2 h, BQ. 1100-1150F for 31 or 40 h, FC[c]
C2800-2	C2800-2	NA	SA-302B, mod. to ASME Code Case 1339	Lukens	1625-1675F for 1 h/in., WQ; 1200-1250F for 1 h/in., AC. Probably also 1600-1650F for 9-1/2 h, BQ; 1200-1225F for 9-1/2 h, BQ. 1100-1150F for 31 or 40 h, FC[c]
WF 112	406L44	Linde 80 8688		Union Carbide	1100-1150F for 31 or 40 h, FC[c]
Correlation plate; HSST 02	A-1195-1				1600F ±75F for 4 h, WQ; 1225F ±25F for 4 h, FC; 1125 ±25F for 40 h, FC

[a]NA denotes not applicable.

[b]FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

[c]Time varies depending on interpretation of weld records.

Table 4.2-4. Unirradiated Mechanical Properties of Plant-Specific RVSP Materials for Oconee Unit 1[16]

		Toughness properties					Tensile properties				
Material ID	Heat treatment	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	YS, ksi	UTS, ksi	El, %	RA, %	Notes
C3265-1	See Table 4.2-3	-10	+15	140--LT			64.3	86.1	26.5	68	Longitudinal
		+10	+65	108--TL			65.1	86.5	26.1	65	Transverse
C2800-2	See Table 4.2-3	+10	+40	118--LT			68.7	89.6	26.8	69	Longitudinal
		+15	+70	115--TL			68.3	89.5	25.4	65	Transverse
HAZ -- C3265-1	See Table 4.2-3						66.5	87.5	28.7	72	Longitudinal
		-75	-15	105--TL			61.6	80.4	24.6	63	Transverse
		-60	-10	100--TS							
HAZ -- C2800-2	See Table 4.2-3						67.7	89.0	26.6	65	Longitudinal
		-60	-12	110--TL			66.9	8.50	20.8	64	Transverse
		<-80	-80	110--TS							
WF 112	See Table 4.2-3	5	+50	64--TL			63.3	80.5	30.9	63	
HSST 02	See Table 4.2-3	+56	+93	130--LT							

Table 4.2-5. RVSP Specimen Orientation Information for Oconee Unit 1

Heat ID	Specimen ID	Orientation
Tensile Specimens		
C3265-1	AA 601-612	Transverse
C3265-1	AA 701-730	Longitudinal
C2800-2	BB 601-612	Transverse
C2800-2	BB 701-730	Longitudinal
HAZ -- C3265-1	AA 301-306	Longitudinal to weld and plate
HAZ -- C3265-1	AA 401-409	Transverse to weld and plate
HAZ -- C2800-2	BB 301-306	Longitudinal to weld and plate
HAZ -- C2800-2	BB 401-427	Transverse to weld and plate
Weld -- WF 112	OCI 101-127	Longitudinal
Charpy Specimens		
C3265-1	AA 601-640+, AA 695-700	TL
C3265-1	AA 701-756+, AA 895-900	LT
C2800-2	BB 601-656+, BB 689-694	TL
C2800-2	BB 701-764+, BB 889-894	LT
HAZ -- C3265-1	AA 301-320, AA 595-599	Transverse to weld, TS to plate
HAZ -- C3265-1	AA 401-452, AA 395-400	Transverse to weld, TL to plate
HAZ -- C2800-2	BB 301-320+, BB 589-594	Transverse to weld, TS to plate
HAZ -- C2800-2	BB 401-464, BB 390-394	Transverse to weld, TL to plate
Weld -- WF 112	OCI 001-060+	Transverse, TL to plate
HSST plate 02	AA 901-975	LT

4.3. Oconee Unit 2 RVSP

Table 4.3-1. Plant-Specific RVSP Information for Oconee Unit 2

B&W Documentation: BAW-10006A, Rev. 3, BAW-1543, Rev. 2, R&D Division Report 9015[7]

Number of Surveillance Capsules: Six

Irradiation Sites: Oconee Unit 2 and Crystal River Unit 3

Surveillance Program Materials:

Base Metal -- AAW 163
AWG 164

HAZ Metal -- AAW 163
AWG 164

Weld Metal -- WF 209-1

Correlation Metal -- HSST Plate 02

Surveillance Program Specimens:

Material	Orientation	Tensile	CVN
Baseline			
AAW 163	Longitudinal	6	27
AAW 163	Transverse	6	27
AWG 164	Longitudinal	6	27
AWG 164	Transverse	6	26
HAZ -- AAW 163	Longitudinal	6	0
HAZ -- AAW 163	Transverse	6	27 (TL); 27 (TS)
HAZ -- AWG 164	Longitudinal	6	0
HAZ -- AWG 164	Transverse	6	27 (TL); 26 (TS)
Weld -- WF 209-1		6	15 (TL)
Capsules A, C, E			
AAW 163	Longitudinal	4	8 (LT)
AAW 163	Transverse	0	4 (TL)
HAZ -- AAW 163	Transverse	0	8 (TL)
Weld -- WF 209-1		4	8 (TL)
HSST Plate 02	Longitudinal	0	8 (LT)

Table 4.3-1. Cont'd)

Material	Orientation	Tensile	CVN
Capsules B, D, F			
AWG 164	Longitudinal	4	10 (LT)
AWG 164	Transverse	0	8 (TL)
HAZ -- AWG 164	Transverse	4	10 (TL)
HSST Plate 02	Longitudinal	0	8 (LT)

Table 4.3-2. Chemical Composition of Plant-Specific RVSP Materials for Oconee Unit 2

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
AAW 163	0.24*	0.63	0.006	0.012	0.25	0.75	0.36	0.62	0.04	0.02 V, 0.01 Co Product analysis
	0.23	0.63	0.006	0.013	0.27	0.78	0.31	0.62	0.04	0.04 V, 0.01 Co Ladle analysis
	0.22	0.66	0.006	0.012	0.27	0.78	0.36	0.58	0.04	0.03 V, 0.01 Co Ladle analysis
AWG 164	0.21*	0.62	0.010	0.010	0.23	0.80	0.39	0.58	0.02	0.01 V, 0.01 Co Product analysis
	0.21	0.66	0.006	0.011	0.27	0.77	0.39	0.61	0.05	0.03 V, 0.01 Co Average of 3 ladle analyses
WF 209-1	0.11*	1.55	0.022	0.010	0.65	0.58	0.09	0.39	0.36	BAW-1500, Table 5
	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	BAW-1500, Table 6
	0.067	1.58	0.020	0.005	0.56	0.48	0.12	0.33	0.30	Weld qualification
Correlation plate; HSST 02	0.23	1.39	0.013	0.013	0.21	0.64	--	0.50	0.17	ORNL-4463

*Used for data analysis.

Table 4.3-3. Heat Treatment of Plant-Specific RVSP Materials for Oconee Unit 2

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
AAW 163	3P2359	NA	A 508 Cl. 2 mod. by ASME Code Case 1332-4	Ladish	1640F ±20F for 4 h, WQ; 1590F ±20F for 4 h, WQ; 1260F ±20F for 10 h, WQ; 1100-1150F for 27 or 33 h, FC
AWG 164	4P1885	NA	A 508 Cl. 2 mod. by ASME Code Case 1332-4	Ladish	1640F ±20F for 4 h, WQ; 1590F ±20F for 4 h, WQ; 1260F ±20F for 10 h, WQ; 1100-1150F for 27 or 33 h, FC
WF 209-1	72105	Linde 80 8773		Amer. Chain & Cable	1100-1150F for 27 or 33 h, FC
Correlation plate, HSST 02	A-1195-1				1600F ± 75F for 4 h, WQ/ 1225F ±25F for 4 h, FC; 1125F ±25F for 40 h, FC

(a)NA denotes not applicable.

(b)FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Table 4.3-4. Unirradiated Mechanical Properties of Plant-Specific RVSP Materials for Oconee Unit 2[17]

		Toughness properties					Tensile properties				
Material ID	Heat treatment	C_V 30, F	C_V 50, F	C_V USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	YS, ksi	UTS, ksi	El, %	RA, %	Notes
AAW 163	See Table 4.2-3	-24	-4	152--LT			68.0	89.2	28.1	69.7	Longitudinal
		-28	-7	133--TL			68.7	89.6	26.6	65.3	Transverse
AWG 164	See Table 4.2-3	-25	-2	160--LT			69.5	89.9	27.7	71.7	Longitudinal
		<0	+9	138--TL			67.1	87.8	27.1	69.1	Transverse
HAZ -- AWG 163	See Table 4.2-3						68.3	89.4	28.3	70.7	Longitudinal
		<-40	<-40	142--TL			70.7	91.7	21.9	65.3	Transverse
		<-40	<+14	126--TS							
HAZ -- AWG 164	See Table 4.2-3						64.3	88.5	20.0	69.0	Longitudinal
		<-40	<-15	125--TL			66.4	87.6	28.8	71.6	Transverse
		<-60	<+16	149--TS							
Weld -- WF 209-1	See Table 4.2-3	+4	+50	67--TL			81.4	95.2	25.6	57.9	
HSST 02	See Table 4.2-3	+56	+93	130--LT							

Table 4.3-5. RVSP Specimen Orientation Information for Oconee Unit 2

Heat ID	Specimen ID	Orientation
Tensile Specimens		
AAW 163	EE 601-606	Transverse
AAW 163	EE 701-718+	Longitudinal
AWG 164	FF 601-606	Transverse
AWG 164	FF 701-718	Longitudinal
HAZ -- AAW 163	EE 301-312	Longitudinal to weld and forging
HAZ -- AAW 163	EE 401-406+	Transverse to weld and forging
HAZ -- AWG 164	FF 301-312	Longitudinal to weld and forging
HAZ -- AWG 164	FF 401-418+	Transverse to weld and forging
Weld -- WF 209-1	EE 101-124	Longitudinal
Charpy Specimens		
AAW 163	EE 601-628+	TL
AAW 163	EE 701+	Assumed LT -- no drawing available
AWG 164	FF 601-640+	TL
AWG 164	FF 701-748+	LT
HAZ -- AAW 163	EE 301-320+	Transverse to weld, TS to forging
HAZ -- AAW 163	EE 401-440+	Transverse to weld, TL to forging
HAZ -- AWG 164	FF 301-316+	Transverse to weld, TS to forging
HAZ -- AWG 164	FF 401-448+	Transverse to weld, TL to forging
Weld -- WF 209-1	EE 001-040+	Transverse to weld, TL to forging
HSST plate 02	EE/FF 901-960	LT

4.4. Three Mile Island Unit 1 RVSP

Table 4.4-1. Plant-Specific RVSP Information for Three Mile Island Unit 1

B&W Documentation: BAW-10006A, Rev. 3; BAW-1543, Rev. 2; R&D Division Report 9007[8]

Number of Surveillance Capsules: Six

Irradiation Sites: Three Mile Island Unit 1, Three Mile Island Unit 2, and Crystal River 3

Surveillance Program Materials:

Base Metal -- C2789-2
C3307-1

HAZ Metal -- C2789-2
C3307-1

Weld Metal -- WF 25

Correlation Metal -- HSST Plate 02

Surveillance Program Specimens:

Material	Orientation	Tensile	CVN
Baseline			
C2789-2	Longitudinal	6	29 (LT)
C2789-2	Transverse	6	27 (TL)
C3307-1	Longitudinal	6	27 (LT)
C3307-1	Transverse	6	27 (TL)
HAZ -- C2789-2	Longitudinal	6	0
HAZ -- C2789-2	Transverse	6	31 (TL); 28 (TS)
HAZ -- C3307-1	Longitudinal	6	0
HAZ -- C3307-1	Transverse	6	27 (TL); 24 (TS)
Weld -- WF 25		6	18 (TL)
Capsules A, C, E			
C2789-2	Longitudinal	4	8 (LT)
C2789-2	Transverse	0	4 (TL)
HAZ -- C2789-2	Transverse	0	8 (TL)
Weld -- WF 25		4	8 (TL)
HSST Plate 02	Longitudinal	0	8 (LT)

Table 4.4-1. (Cont'd)

Material	Orientation	Tensile	CVN
Capsules B, D, F			
C3307-1	Longitudinal	4	10 (LT)
C3307-1	Transverse	0	8 (TL)
HAZ -- C3307-1	Transverse	4	10 (TL)
HSST Plate 02	Longitudinal	0	8 (LT)

Table 4.4-2. Chemical Composition of Plant-Specific RVSP Materials for Three Mile Island Unit 1

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
C2789-2	0.24	1.36	0.010	0.017	0.23	0.57	0.19	0.51	0.09	0.015 Co
C3307-1	0.21	1.24	0.010	0.016	0.27	0.55	0.12	0.47	0.12	0.015 Co
WF 25	0.09*	1.62	0.014	0.015	0.46	0.66	0.10	0.40	0.33	BAW-1500, Table 5
	0.09	1.60	0.015	0.016	0.50	0.68	0.09	0.42	0.35	BAW-1500, Table 10
	0.09	1.58	0.014	0.016	0.53	0.68	0.09	0.42	0.35	BAW-1500, Table 6
	0.088	1.50	0.019	0.010	0.45	0.71	0.11	0.33	0.29	Weld qualification
Correlation plate; HSST 02	0.23	1.39	0.013	0.013	0.21	0.64	--	0.50	0.17	ORNL-4463

*Used for data analysis.

Table 4.4-3. Heat Treatment of Plant-Specific RVSP Materials for Three Mile Island Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
C2789-2	C2789-2	NA	SA-302 Gr. B, mod. by ASME Code Case 1339	Lukens	Supplier heat treatment unknown. 1600-1650F for 9-1/2 h, BQ; 1200-1225F for 9-1/2 h, BQ; 1600-1650F for 9-1/2 h, BQ; 1600-1650F for 9-1/2 h, BQ; 1510-1535F for 5 h, BQ; 1200-1225F for 5 h, BQ; 1100-1150F for 22-1/2 h or 27-1/2 h, FC[c]
C3307-1	C3307-1	NA	SA-302 Gr. B, mod. by ASME Code Case 1339	Lukens	Supplier heat treatment unknown. 1600-1650F for 9-1/2 h, BQ; 1200-1225F for 9-1/2 h, BQ; 1225-1250F for 9-1/2 h, BQ; 1100-1150F for 22-1/2 h or 27-1/2 h, FC[c]
WF 25	299L44	Linde 80 8650		Union Carbide	1100-1150F for 22-1/2 h or 27-1/2 h, FC[c]
HSST 02	A-1195-1		A 533 Gr. B, Cl. 1		1600F ±75F for 4 h, WQ; 1225F ±25F for 4 h, FC; 1125 ±25F for 40 h, FC

(a) NA denotes not applicable.

(b) FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

(c) Time varies depending on interpretation of weld records.

Table 4.4-4. Unirradiated Mechanical Properties of Plant-Specific RVSP Materials for Three Mile Island Unit 1[18]

Material ID	Heat treatment	Toughness properties: C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile properties: YS, ksi	UTS, ksi	El, %	RA, %	Notes
C2789-2	See Table 4.4-3	-20	+26	131--LT			71.1	94.0	26.6	68.0	Longitudinal
		+30	+70	98--TL			68.4	92.2	24.0	63.1	Transverse
C3307-1	See Table 4.4-3	0	+38	182--LT			60.4	83.2	31.2	70.8	Longitudinal
		+10	+73	112--TL			59.4	82.4	29.3	60.4	Transverse
HAZ -- C2789-2	See Table 4.4-3						69.7	94.1	27.1	62.6	Longitudinal
		<-50	<-38	132--TL			65.4	83.2	19.3	63.4	Transverse
		<-60	<-15	108--TS							
HAZ -- C3307-1	See Table 4.4-3						52.1	86.2	30.2	65.6	Longitudinal
		<-60	<-40	>200--TL			58.9	83.1	23.4	70.5	Transverse
		<-50	<-10	150--TS							
WF 25	See Table 4.4-3	<0	+39	81--TL			69.2	86.2	26.7	62.8	
HSST 02	See Table 4.4-3	+56	+93	130--LT							

Table 4.4-5. RVSP Specimen Orientation Information for Three Mile Island Unit 1

Heat ID	Specimen ID	Orientation
Tensile Specimens		
C2789-2	CC 601-612	Transverse
C2789-2	CC 701-727	Longitudinal
C3307-1	DD 601-612	Transverse
C3307-1	DD 701-727	Longitudinal
HAZ -- C2789-2	CC 301-306	Longitudinal to plate and weld
HAZ -- C2789-2	CC 401-409	Transverse to plate and weld
HAZ -- C3307-1	DD 301-306	Longitudinal to plate and weld
HAZ -- C3307-1	DD 401-424	Transverse to plate and weld
WELD -- WF 25	CC 101-127	Longitudinal
Charpy Specimens		
C2789-2	CC 601-656+	TL
C2789-2	CC 701-764+	LT
CC307-1	DD 601-656+	TL
C3307-1	DD 701-764+	LT
HAZ -- C2789-2	CC 301-328+	Transverse to weld, TS to plate
HAZ -- C2789-2	CC 401-452+	Transverse to weld, TL to plate
HAZ -- C3307-1	DD 301-320+	Transverse to weld, TS to plate
HAZ -- C3307-1	DD 401-460+	Transverse to weld, TL to plate
Weld -- WF 25	CC 001-060+	Transverse, TL to plate
HSST plate 02	CC/DD 901-970	LT

4.5. Crystal River Unit 3 RVSP

Table 4.5-1. Plant-Specific RVSP Information for Crystal River Unit 3

B&W Documentation: BAW-10100A[19]; BAW-1543, Rev. 2; LRC 9060[9]

Number of Surveillance Capsules: Six

Irradiation Site: Crystal River Unit 3

Surveillance Program Materials:

Base Metal -- C4344-1
C4344-2

HAZ Metal -- C4344-1
C4344-2

Weld Metal -- WF 209-1

Weld -- atypical

Correlation Metal -- HSST Plate 02

Surveillance Program Specimens:

Material	Orientation	Tensile	CVN	0.5TCT	1.0TCT
Baseline					
C4344-1	Longitudinal	6	15 (LT)		
C4344-1	Transverse	6	15 (TL)		
C4344-2	Longitudinal	6	15 (LT)		
C4344-2	Transverse	6	15 (TL)		
HAZ -- C4344-1	Longitudinal	6	0		
HAZ -- C4344-1	Transverse	6	15 (TL); 15 (TS)		
HAZ -- C4344-2	Longitudinal	6	0		
HAZ -- C4344-2	Transverse	6	15 (TL); 15 (TS)		
Weld -- WF 209-1				8 (TL)	6 (TL)
Weld -- atypical		6	15 (TL)		
Prob. WF 209-1				1 (TL)	
Prob. C4344-1				7 (TL)	
Capsules A, C, E					
C4344-1	Transverse	2	12 (TL)		
C4344-2	Transverse	0	6		
HAZ -- C4344-1	Transverse	0	12 (TL)		

Table 4.5-1. (Cont'd)

Material	Orientation	Tensile	CVN	0.5TCT	1.0TCT
Capsules A, C, E (Cont'd)					
HAZ -- C4344-2	Transverse	0	6 (TL)		
Weld -- atypical		2	12 (TL)		
HSST Plate 02	Longitudinal	0	6 (LT)		
Weld -- atypical		2	12 (TL)		
HSST plate 02	Longitudinal	0	6 (LT)		
Capsules B, D, F					
C4344-1	Transverse	2	12	(a)	
HAZ -- C4344-1	Transverse	0	12 (TL)		
Weld -- atypical		2	12 (TL)		
Weld -- WF 209-1				(a)	

[a] Capsule B contains three WF 209-1 CT specimens, three base metal CT specimens which are probably C4344-1, and two CT specimens expected to be WF 209-1. Capsule D contains four WF 209-1 CT specimens, three base metal CT specimens which are probably C4344-1, and one CT specimen expected to be base metal, probably C4344-1. Capsule F contains five WF 209-1 CT specimens, one base metal CT specimen which is probably C4344-1, one CT specimen expected to be WF 209-1, and one CT specimen expected to be base metal, probably C4344-1.

Table 4.5-2. Chemical Composition of Plant-Specific RVSP Materials for Crystal River Unit 3

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
C4344-1	0.23	1.30	0.008	0.016	0.22	0.54	0.11	0.55	0.20	0.013 Co
C4344-2	0.23	1.30	0.008	0.016	0.22	0.54	0.11	0.55	0.20	0.013 Co
WF 209-1	0.11*	1.57	0.018	0.009	0.54	0.61	0.09	0.43	0.36	BAW-1500, Table 5
	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	BAW-1500, Table 6
	0.067	1.58	0.020	0.005	0.56	0.48	0.12	0.33	0.30	Weld qualification
Atypical weld	0.08	1.65	0.021	0.013	1.00	0.10	0.073	0.45	0.39	BAW-10144[20]
Correlation plate; HSST 02	0.23	1.39	0.013	0.013	0.21	0.64	--	0.50	0.17	ORNL-4463

*Used for data analysis.

Table 4.5-3. Heat Treatment of Plant-Specific RVSP Materials for Crystal River Unit 3

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
C4344-1	C4344-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1180F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1175-1200F for 6 h, BQ. 1100-1150F for 27 h, FC
C4344-2	C4344-2	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1100F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1175-1200F for 6 h, BQ. 1100-1150F for 27 h, FC
WF 209-1	72105	Linde 80 8773		Amer. Chain & Cable	1100-1150F for 27 h
Atypical weld	Unknown	Linde 80 8773		Amer. Chain & Cable	1100-1150F for 27 h
Corelation plate; HSST 02	A-1195-1				1600F ±75F for 4 h, WQ; 1225F ±25F for 4 h, FC; 1125F ±25F for 40 h, FC

(a)NA denotes not applicable.

(b)FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Table 4.5-4. Unirradiated Mechanical Properties of Plant-Specific RVSP Materials for Crystal River Unit 3[21]

Material ID	Heat treatment	Toughness properties					Tensile properties				Notes
		C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	YS, ksi	UTS, ksi	El, %	RA, %	
C4344-1	See Table 4.5-3	<-20	+14	124--LT			70.0	91.9	26.6	69.3	Longitudinal
		+16	+61	94--TL			69.4	92.3	24.[illegible]	62.3	Transverse
C4344-2[a]	See Table 4.5-3			--LT							Longitudinal
				--TL							Transverse
HAZ -- C4344-1	See Table 4.5-3						69.0	91.5	28.3	69.3	Longitudinal
		-19	+50	84--TL			69.4	91.9	22.3	63.3	Transverse
		-72	+28	95--TS							
HAZ -- C4344-2(2)	See Table 4.5-3										Longitudinal
				--TL							Transverse
				--TS							
WF 209-1											
Atypical weld	See Table 4.5-3	+36	+112	79--TL			77.1	93.9	29.0	62.6	
HSST 02	See Table 4.5-3	+56	+93	130--LT							

(a)No surveillance specimens were tested at this time.

Table 4.5-5. RVSP Specimen Orientation Information for Crystal River Unit 3

Heat ID	Specimen ID	Orientation
Tensile Specimens		
C4344-1	NN 601-621	Transverse
C4344-1	NN 801-806	Longitudinal
C4344-2	PP 601-606	Transverse
C4344-2	PP 801-806	Longitudinal
HAZ -- C4344-1	NN 301-306	Transverse to plate and weld
HAZ -- C4344-1	NN 501-509	Longitudinal to plate and weld
HAZ -- C4344-2	NN 301-306	Transverse to plate and weld
HAZ -- C4344-2	NN 501-507	Longitudinal to plate and weld
Weld -- atypical	PP 001-019	Longitudinal
Charpy Specimens		
C4344-1	NN 601-692+	TL
C4344-1	NN 801-820+	LT
C4344-2	PP 601-644	TL
C4344-2	PP 801-821	LT
HAZ -- C4344-1	NN 301-392+	Transverse to weld, TL to plate
HAZ -- C4344-1	NN 501-516+	Transverse to weld, TS to plate
HAZ -- C4344-2	PP 301-340+	Transverse to weld, TL to plate
HAZ -- C4344-2	PP 501-516	Transverse to weld, TS to plate
Weld -- atypical	PP 001-092+	Transverse, TL to plate
HSST plate 02	NN 901-920+	LT
1TCT Specimens		
Weld -- WF 209-1	NN 001-006	Transverse, TL
0.5TCT Specimens		
Base metal (prob. C4344-1)	NN 021-032	Probably TL
Prob. base metal (C4344-1)	NN 033-036	Probably TL
Weld -- WF 209-1	NN 001-020	Transverse, TL
Prob. Weld -- WF 209-1	NN 037-040	Probably transverse, TL

4.6. Arkansas Nuclear One Unit 1 RVSP

Table 4.6-1. Plant-Specific RVSP Information for Arkansas Nuclear One Unit 1

B&W Documentation: BAW-10006A, Rev. 3; BAW-1543, Rev. 2; R&D Division Report 9040[10]

Number of Surveillance Capsules: Six

Irradiation Sites: Arkansas Nuclear One Unit 1 and Davis-Besse Unit 1

Surveillance Program Materials:

Base Metal -- C5114-1
C5114-2

HAZ Metal -- C5114-1
C5114-2

Weld Metal -- WF 193

Correlation Metal -- HSST Plate 02

Surveillance Program Specimens:

Material	Orientation	Tensile	CVN
Baseline			
C5114-1	Longitudinal	6	27
C5114-1	Transverse	6	26
C5114-2	Longitudinal	6	27
C5114-2	Transverse	6	27
HAZ -- C5114-1	Longitudinal	5	0
HAZ -- C5114-1	Transverse	6	26 (TL); 27 (TS)
HAZ -- C5114-2	Longitudinal	6	0
HAZ -- C5114-2	Transverse	6	26 (TL); 27 (TS)
Weld -- WF 193		6	27 (TL)
Capsules A, C, E			
C5114-1	Longitudinal	4	8[(a)] (LT)
C5114-1	Transverse	0	4[(a)] (TL)
HAZ -- C5114-1	Transverse	0	8 (TL)
Weld -- WF 193		4	8 (TL)
HSST plate 02	Longitudinal	0	8 (LT)

Table 4.6-1. (Cont'd)

Material	Orientation	Tensile	CVN
Capsules B, D, F			
C5114-2	Longitudinal	4	10 (LT)
C5114-2	Transverse	0	8 (TL)
HAZ -- C5114-2	Transverse	4	10 (TL)
HSST plate 02	Longitudinal	0	8 (LT)

[(a)]Except capsule A, which contained seven LT and five TL CVN specimens.

Table 4.6-2. Chemical Composition of Plant-Specific RVSP Materials for Arkansas Nuclear One Unit 1

	Chemical Composition, wt %									
Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Notes
C5114-1	0.21	1.32	0.010	0.016	0.20	0.52	0.19	0 57	0.15	0.012 Co
C5114-2	0.21	1.32	0.010	0.016	0.20	0.52	0.19	0.57	0.15	0.012 Co
WF 193	0.09*	1.49	0.016	0.016	0.51	0.59	0.06	0.39	0.28	BAW-1500, Table 5
	0.09	1.49	0.016	0.016	0.52	0.59	0.07	0.40	0.31	BAW-1500, Table 10
	0.08	1.48	0.016	0.016	0.53	0.59	0.06	0.40	0.31	BAW-1500, Table 6
	0.065	1.50	0.016	0.008	0.42	0.59	--	0.36	0.19	Weld qualification
Correlation plate; HSST 02	0.23	1.39	0.013	0.013	0.21	0.64	--	0.50	0.17	ORNL-4463

*Used for data analysis.

Table 4.6-3. Heat Treatment of Plant-Specific RVSP Materials for Arkansas Nuclear One Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
C5114-1	C5114-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ. 1100-1150F for 29 h, FC
C5114-2	C5114-2	NA	SA-533 Gr. B Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ. 1100-1150F for 29 h, FC
WF 193	406L44	Linde 80 8773		Union Carbide	1100-1150F for 29 h, FC
Correlation plate; HSST 02	A-1195-1				1600F ±75F for 4 h, WQ; 1225F ±25F for 4 h, FC; 1125 ±25F for 40 h, FC

[a] NA denotes not applicable.

[b] FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Table 4.6- Unirradiated Mechanical Properties of Plant-Specific RVSP Materials for Arkansas Nuclear One Unit 1[22]

Material ID	Heat treatment	Toughness properties: C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	Tensile properties: YS, ksi	UTS, ksi	El, %	RA, %	Notes
C5114-1	See Table 4.6-3	<-40	+8	132--LT			72.0	94.9	26.7	68.2	Longitudinal
		+7	+74	107--TL			71.8	94.6	24.3	60.6	Transverse
C5114-2	See Table 4.6-3	-10	+18	147--LT			67.7	90.1	28.1	68.8	Longitudinal
		+20	+52	107--TL			67.8	90.3	27.1	64.8	Transverse
HAZ -- C5114-1	See Table 4.6-3						71.1	94.2	27.4	68.8	Longitudinal
		<-40	0	115--TL			69.7	87.3	18.3	63.6	Transverse
		<-40	0	95--TS							
HAZ -- C5114-2	See Table 4.6-3						65.0	88.7	28.6	71.7	Longitudinal
		<-40	+4	110--TL			65.6	85.6	22.4	63.8	Transverse
		<-40	-20	104--TS							
Weld -- WF 193	See Table 4.6-3	+5	+65	73--TL			67.6	84.6	28.1	64.0	
HSST 02	See Table 4.6-3	+56	+93	130--LT							

Table 4.6-5. RVSP Specimen Orientation Information for Arkansas Nuclear One Unit 1

Heat ID	Specimen ID	Orientation
Tensile Specimens		
C5114-1	GG 601-606	Transverse
C5114-1	GG 701-718	Longitudinal
C5114-2	HH 601-609	Transverse
C5114-2	HH 701-721	Longitudinal
HAZ -- C5114-1	GG 301-306	Longitudinal to plate and weld
HAZ -- C5114-1	GG 401-406	Transverse to plate and weld
HAZ -- C5114-2	HH 301-306	Longitudinal to plate and weld
HAZ -- C5114-2	HH 401-406	Transverse to plate and weld
Weld -- WF 193	GG 101-118	Longitudinal to weld
Charpy Specimens		
C5114-1	GG 601-628+	TL
C5114-1	GG 701-744+	LT
C5114-2	HH 601-644+	TL
C5114-2	HH 701-748+	LT
HAZ -- C5114-1	GG 301-320+	Transverse to weld, TS to plate
HAZ -- C5114-1	GG 401-444+	Transverse to weld, TL to plate
HAZ -- C5114-2	HH 301-316+	Transverse to weld, TS to plate
HAZ -- C5114-2	HH 401-448+	Transverse to weld, TL to plate
Weld -- WF 193	GG 001-044+	Transverse, TL to plate
HSST plate J2	GG/HH 901-950	LT

4.7. Oconee Unit 3 RVSP

Table 4.7-1. Plant-Specific RVSP Information for Oconee Unit 3

B&W Documentation: BAW-10006A, Rev. 3; BAW-1543, Rev. 2; R&D Division Report 9025[11]

Number of Surveillance Capsules: Six

Irradiation Sites: Oconee Unit 3 and Crystal River Unit 3

Surveillance Program Materials:

Base Metal -- ANK 191
AWS 192

HAZ Metal -- ANK 191
AWS 192

Weld Metal -- WF 209-1

Correlation Metal -- HSST Plate 02

Surveillance Program Specimens:

Material	Orientation	Tensile	CVN
Baseline			
ANK 191	Longitudinal	6	30 (LT)
ANK 191	Transverse	6	29 (TL)
AWS 192	Longitudinal	6	27 (LT)
AWS 192	Transverse	6	29 (TL)
HAZ -- ANK 191	Longitudinal	6	0
HAZ -- ANK 191	Transverse	6	21 (TL); 27 (TS)
HAZ -- AWS 192	Longitudinal	6	0
HAZ -- AWS 192	Transverse	6	29 (TL); 27 (TS)
Weld -- WF 209-1		6	21 (TL)
Capsules A, C, E			
ANK 191	Longitudinal	0	9 (LT)
ANK 191	Transverse	2	12 (TL)
AWS 192	Transverse	0	9 (TL)
HAZ -- ANK 191	Transverse	0	12 (TL)
Weld -- WF 209-1		2	12 (TL)

Table 4.7-1. (Cont'd)

Material	Orientation	Tensile	CVN
Capsules B, D, F			
ANK 191	Transverse	2	12 (TL)
AWS 192	Transverse	0	6 (TL)
HAZ -- ANK 191	Transverse	0	12 (TL)
HAZ -- AWS 192	Transverse	0	6 (TL)
Weld -- WF 209-1		2	12 (TL)
HSST plate 02	Longitudinal	0	6 (LT)

Table 4.7-2. Chemical Composition of Plant-Specific RVSP Materials for Oconee Unit 3

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
ANK 191	0.27	0.77	0.013	0.010	0.23	0.75	0.34	0.57	0.04	<0.01 V, 0.01 Co Ladle analysis
	0.27	0.76	0.012	0.010	0.25	0.75	0.34	0.57	0.04	<0.01 V, 0.01 Co Ladle analysis
	0.24*	0.72	0.014	0.012	0.21	0.76	0.34	0.62	0.02	<0.01 V, 0.01 Co Product analysis
AWS 192	0.22	0.65	0.008	0.011	0.22	0.73	0.30	0.56	0.04	0.007 V, 0.01 Co Ladle analysis
	0.22	0.64	0.009	0.014	0.21	0.73	0.32	0.57	0.04	0.007 V, 0.01 Co Ladle analysis
	0.21*	0.58	0.011	0.015	0.24	0.73	0.30	0.60	0.01	<0.01 V, 0.01 Co Product analysis
WF 209-1	0.08*	1.63	0.017	0.012	0.61	0.58	0.10	0.39	0.30	BAW-1500, Table 5
	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	BAW-1500, Table 6
	0.067	1.58	0.020	0.005	0.56	0.48	0.12	0.3[illegible]	0.30	Weld qualification
Correlation plate; HSST 02	0.23	1.39	0.013	0.013	0.21	0.64	--	0.50	0.17	ORNL-4463

*Used for data analysis.

Table 4.7-3. Heat Treatment of Plant-Specific RVSP Materials for Oconee Unit 3

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
ANK 191	522194	NA	A 508-64 Cl. 2, mod. to ASME Code Case 1332-3	Ladish	1640F ±20F for 4 h, WQ; 1590F ±20F for 4 h, WQ; 1250F ±20F for 10 h, WQ; 1100-1150F for 30 h, FC
AWS 192	522314	NA	A 508-64 Cl. 2, mod. to ASME Code Case 1332-3	Ladish	1640F ±20F for 4 h, WQ; 1590F ±20F for 4 h, WQ; 1240F ±20F for 10 h, WQ; 1100-1150F for 30 h, FC
WF 209-1	72105	Linde 80 8773		Amer. Chain & Cable	1100-1150F for 30 h, FC
Correlation plate; HSST 02	A-1195-1				1600F ±75F for 4 h, WQ; 1225F ±25F for 4 h, FC; 1125F ±25F for 40 h, FC

[a]NA denotes not applicable.

[b]FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Table 4.7-4. Unirradiated Mechanical Properties of Plant-Specific RVSP Materials for Oconee Unit 3[23]

Material ID	Heat treatment	Toughness properties					Tensile properties				Notes
		C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	YS, ksi	UTS, ksi	El, %	RA, %	
ANK 191	See Table 4.7-3	-9	+12	148--LT			63.1	85.4	30.4	66.8	Longitudinal
		+8	+23	180--TL			59.1	84.0	30.2	71.2	Transverse
AWS 192	See Table 4.7-3	-9	+37	112--LT			58.2	83.1	27.6	64.1	Longitudinal
		<-40	-12	160--TL			59.6	84.6	28.4	68.9	Transverse
HAZ -- ANK 191	See Table 4.7-3						59.0	84.2	30.2	70.7	Longitudinal
		-30	+32	92--TL			59.0	84.6	21.9	63.7	Transverse
		<0	<+80	>65--TS							
HAZ -- AWS 192	See Table 4.7-3						55.7	81.7	29.8	69.5	Longitudinal
		-56	-32	125--TL			56.6	83.2	24.3	66.3	Transverse
		-70	-44	140--TS							
Weld -- WF 209-1	See Table 4.7-3	+45	+85	~63--TL			75.0	90.5	28.1	62.9	
HSST 02	See Table 4.7-3	+56	+93	130--LT							

Table 4.7-5. RVSP Specimen Orientation Information for Oconee Unit 3

Heat ID	Specimen ID	Orientation
Tensile Specimens		
ANK 191	JJ 601-621	Transverse
ANK 191	JJ 801-806	Longitudinal
AWS 192	KK 601-609	Transverse
AWS 192	KK 801-806	Longitudinal
HAZ -- ANK 191	JJ 301-306	Transverse to weld and forging
HAZ -- ANK 191	JJ 501-509	Longitudinal to weld and forging
HAZ -- AWS 192	KK 301-306	Transverse to weld and forging
HAZ -- AWS 192	KK 501-506	Longitudinal to weld and forging
Weld -- WF 209-1	JJ 001-018	Longitudinal
Charpy Specimens		
ANK 191	JJ 601-696+	TL
ANK 191	JJ 801-848+	LT
AWS 192	KK 601-672+	TL
AWS 192	KK 801-816+	LT
HAZ -- ANK 191	JJ 301-416+	Transverse to weld, TL to forging
HAZ -- ANK 191	JJ 501-516+	Transverse to weld, TS to forging
HAZ -- AWS 192	KK 301-336+	Transverse to weld, TL to forging
HAZ -- AWS 192	KK 501-516+	Transverse to weld, TS to forging
Weld -- WF 209-1	JJ 001-096	Transverse, TL to forging
HSST plate 02	JJ 901-918	LT

4.8. Rancho Seco Unit 1 RVSP

Table 4.8-1. Plant-Specific RVSP Information for Rancho Seco Unit 1

B&W Documentation: BAW-10100A; BAW-1543, Rev. 2; LRC 9042[12]

Number of Surveillance Capsules: Six

Irradiation Sites: Rancho Seco Unit 1 and Davis-Besse Unit 1

Surveillance Program Materials:

Base Metal -- C5062-1
C5070-1

HAZ Metal -- C5062-1
C5070-1

Weld Metal -- WF 193

Correlation Metal -- HSST Plate 02

Surveillance Program Specimens:

Material	Orientation	Tensile	CVN	0.5 TCT
Baseline				
C5062-1	Longitudinal	6	15 (LT)	
C5062-1	Transverse	6	14 (TL)	
C5070-1	Longitudinal	6	15 (LT)	
C5070-1	Transverse	6	15 (TL)	
HAZ -- C5062-1	Longitudinal	6	0	
HAZ -- C5062-1	Transverse	6	15 (TL); 15 (TS)	
HAZ -- C5070-1	Longitudinal	6	0	
HAZ -- C5070-1	Transverse	6	15 (TL); 15 (TS)	
Weld -- WF 193		5	14 (TL)	8 (TL)
Capsules A, C, E				
C5062-1	Transverse	2	12 (TL)	
C5070-1	Transverse	0	6 (TL)	
HAZ -- C5062-1	Transverse	0	12 (TL)	
HAZ -- C5070-1	Transverse	0	12	
Weld -- WF 193		2	12	
HSST Plate 02	Longitudinal	0	6 (LT)	

Table 4.8-1. (Cont'd)

Material	Orientation	Tensile	CVN	0.5 TCT
Capsules B, D, F				
C5062-1	Transverse	2	12 (TL)	
HAZ -- C5062-1	Transverse	0	12 (TL)	
Weld -- WF 193		2	12 (TL)	8 (TL)

Table 4.8-2. Chemical Composition of Plant-Specific RVSP Materials for Rancho Seco Unit 1

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
C5062-1	0.20	1.26	0.013	0.017	0.15	0.60	0.14	0.55	0.12	0.007 Co
C5070-1	0.20	1.33	0.010	0.015	0.19	0.58	0.20	0.52	0.10	0.019 Co
WF 193	0.09*	1.49	0.016	0.016	0.52	0.59	0.07	0.40	0.31	BAW-1500, Table 10
	0.08	1.48	0.016	0.016	0.53	0.59	0.06	0.40	0.31	BAW-1500, Table 6
	0.065	1.50	0.016	0.008	0.42	0.59	--	0.36	0.19	Weld qualification
Correlation plate; HSST 02	0.23	1.39	0.013	0.013	0.21	0.64	--	0.50	0.17	ORNL-4463

*Used for data analysis.

Table 4.8-3. Heat Treatment of Plant-Specific RVSP Materials for Rancho Seco Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
C5062-1	C5062-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1220-1225F for 5 h, BQ. 1100-1150F for 28 h, FC
C5070-1	C5070-1	NA	SA-533 Gr. B, Cl. 1	Lukens	1650-1700F for 1 h/in., WQ; 1200F for 1/2 h/in., AC. Probably also 1550-1600F for 4-1/2 h, BQ; 1200-1225F for 5 h, BQ. 1100-1150F for 28 h, FC
WF 193	406L44	Linde 80 8773		Union Carbide	1100-1150F for 28 h, FC
HSST plate 02	A-1195-1				1600F ±75F for 4 h, WQ; 1225F ±25F for 4 h, FC; 1125 ±25F for 40 h, FC

(a) NA denotes not applicable.

(b) FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Table 4.8-4. Unirradiated Mechanical Properties of Plant-Specific RVSP Materials for Rancho Seco Unit 1[24]

		Toughness properties					Tensile properties				
Material ID	Heat treatment	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	YS, ksi	UTS, ksi	El, %	RA, %	Notes
C5062-1	See Table 4.8-3	-30	-5	145--LT			63.8	83.8	29.1	71.0	Longitudinal
		+4	+45	90--TL			63.9	83.8	27.3	66.0	Transverse
C5070-1[a]	See Table 4.8-3			--LT							Longitudinal
				--TL							Transverse
HAZ -- C5062-1	See Table 4.8-3						63.2	83.6	28.9	70.8	Longitudinal
		-52	-19	96--TL			62.7	83.5	20.3	64.5	Transverse
		-30	-10	98--TS							
HAZ --	See Table 4.8-3										Longitudinal
C5070-1[a]				--TL							Transverse
				--TS							
WF 193	See Table 4.8-3	-14	+36	68--TL			67.5	83.4	29.0	63.0	
HSST 02	See Table 4.8-3	+56	+93	130--LT							

[a]No surveillance specimens were tested at this time.

Table 4.8-5. RVSP Specimen Orientation Information for Rancho Seco Unit 1

Heat ID	Specimen ID	Orientation
Tensile Specimens		
C5062-1	LL 601-624	Transverse
C5062-1	LL 801-806	Longitudinal
C5070-1	MM 601-606	Transverse
C5070-1	MM 801-806	Longitudinal
HAZ -- C5062-1	LL 301-306	Transverse
HAZ -- C5062-1	LL 501-506	Longitudinal
HAZ -- C5070-1	MM 301-306	Transverse
HAZ -- C5070-1	MM 501-506	Longitudinal
Weld -- WF 193	MM 001-018	Longitudinal
Charpy Specimens		
C5062-1	LL 601-712+	TL
C5062-1	LL 801-845+	LT
C5070-1	MM 601-680+	TL
C5070-1	MM 801-816+	LT
HAZ -- C5062-1	LL 301-392+	Transverse, TL
HAZ -- C5062-1	LL 501-516	Transverse, TS
HAZ -- C5070-1	MM 301-336+	Transverse, TL
HAZ -- C5070-1	MM 501-516	Transverse, TS
Weld -- WF 193	MM 001-092+	Transverse, TL
HSST plate 02	LL 901-921	LT
0.5 TCT Specimens		
Weld -- WF 193	MM 001-040	Transverse, TL

4.9. Davis-Besse Unit 1 RVSP

Table 4.9-1. Plant-Specific RVSP Information for Davis-Besse Unit 1

B&W Documentation: BAW-10100; BAW-1543, Rev. 2; LRC 9062[13]

Number of Surveillance Capsules: Six

Irradiation Site: Davis-Besse Unit 1

Surveillance Program Materials:

Base Metal -- BCC 241
AKJ 233

HAZ Metal -- BCC 241
AKJ 233

Weld Metal -- WF 182-1

Correlation Metal -- HSST Plate 02

Surveillance Program Specimens:

Material	Orientation	Tensile	CVN	0.5 TCT	1.0 TCT
Baseline					
BCC 241	Longitudinal	6	20 (LT)		
BCC 241	Transverse	6	19 (TL)		
AKJ 233	Longitudinal	6	15 (LT)		
AKJ 233	Transverse	6	15 (TL)		
HAZ -- BCC 241	Longitudinal	6	0		
HAZ -- BCC 241	Transverse	6	20 (TL); 16 (TS)		
HAZ -- AKJ 233	Longitudinal	6	0		
HAZ -- AKJ 233	Transverse	6	15 (TL); 15 (TS)		
Weld -- WF 182-1		5	19 (TL)	8 (TL)	4 (TL)
Capsules A, C, E					
BCC 241	Transverse	2	12 (TL)		
AKJ 233	Transverse	0	6 (TL)		
HAZ -- BCC 241	Transverse	0	12 (TL)		
HAZ -- AKJ 233	Transverse	0	6 (TL)		
Weld -- WF 182-1		2	12 (TL)		
HSST plate 02	Longitudinal	0	6 (LT)		

Table 4.9-1. (Cont'd)

Material	Orientation	Tensile	CVN	0.5 TCT	1.0 TCT
Capsules B, D, F					
BCC 241	Transverse	2	12 (TL)		
HAZ -- BCC 241	Transverse	0	12 (TL)		
Weld -- WF 182-1		2	12 (TL)	8 (TL)	

Table 4.9-2. Chemical Composition of Plant-Specific RVSP Materials for Davis-Besse Unit 1

Material ID	Chemical Composition, wt %									Notes
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	
BCC 241	0.22	0.64	0.012	0.016	0.28	0.83	0.31	0.59	0.04	0.03 V, 0.01 Co Ladle analysis
	0.21	0.63	0.007	0.010	0.25	0.76	0.35	0.63	0.04	0.03 V, 0.01 Co Ladle analysis
	0.22*	0.63	0.011	0.011	0.27	0.81	0.32	0.63	0.02	0.02 V, 0.01 Co Product analysis
AKJ 233	0.24	0.73	0.007	0.006	0.28	0.73	0.37	0.61	--	0.04 V, 0.006 Co Ladle analysis
	0.26*	0.68	0.004	0.006	0.30	0.77	0.38	0.64	0.04	0.04V, 0.01 Co Product analysis
WF 182-1	0.09*	1.70	0.014	0.013	0.42	0.63	0.15	0.40	0.21	BAW-1500, Table 5
	0.08	1.69	0.014	0.013	0.45	0.63	0.14	0.40	0.24	BAW-1500, Table 10
	0.08	1.69	0.014	0.013	0.45	0.63	0.14	0.40	0.24	BAW-1500, Table 6
	0.071	1.60	0.014	0.015	0.30	0.59	--	0.42	0.18	Weld qualification
Correlation plate; HSST 02	0.23	1.39	0.013	0.013	0.21	0.64	--	0.50	0.17	ORNL-4463

*Used for data analysis.

Table 4.9-3. Heat Treatment of Plant-Specific RVSP Materials for Davis-Besse Unit 1

Material ID	Heat ID	Flux ID[a]	Specification	Supplier	Heat Treatment[b]
BCC 241	5P4086	NA	A 508-64 Cl. 2, mod. to ASME Code Case 1332-4	Ladish	1640 ±10F for 4 h, WQ; 1590F ±10F for 4 h, WQ; 1240F ±10F for 5 h, AC; 1100-1150F for 15-1/2 h, FC
AKJ 233	123X244	NA	A 508-64 Cl. 2, mod. to ASME Code Case 1332-4	Ladish	1640F ±10F for 4 h, WQ; 1590F ±10F for 4 h, WQ; 1240F ±10F for 6 h, AC; 1100-1150F for 15-1/2 h, FC
WF 182-1	821T44	Linde 80 8754		Union Carbide Linde	1100-1150F for 15-1/2 h, FC
HSST plate 02	A-1195-1				1600F ±75F for 4 h, WQ; 1225F ±25F for 4 h, FC; 1125F ±25F for 40 h, FC

(a)NA denotes not applicable.

(b)FC denotes furnace cool; AC, air cool; WQ, water quench; and BQ, brine quench.

Table 4.9-4. Unirradiated Mechanical Properties of Plant-Specific RVSP Materials for Davis-Besse Unit 1[25]

		Toughness properties					Tensile properties				
Material ID	Heat treatment	C_v 30, F	C_v 50, F	C_v USE, ft-lbs	T_{NDT}, F	RT_{NDT}, F	YS, ksi	UTS, ksi	El, %	RA, %	Notes
BCC 241	See Table 4.9-3	-25	-10	135--LT			71.7	92.2	26.3	69.4	Longitudinal
		+16	+25	127--TL			72.3	90.7	27.7	68.5	Transverse
AKJ 233[a]	See Table 4.9-3			--LT							Longitudinal
				--TL							Transverse
HAZ -- BCC 241	See Table 4.9-3						72.6	92.9	27.1	68.6	Longitudinal
		-100	-57	130--TL			69.8	90.8	20.1	61.4	Transverse
		-85	-45	120--TS							
HAZ -- AKJ 233[a]	See Table 4.9-3										Longitudinal
				--TL							Transverse
				--TS							
WF 182-1	See Table 4.9-3	-11	+65	70--TL			70.2	85.6	26.6	64.2	
HSST 02	See Table 4.9-3	+56	+93	130--LT							

[a] No surveillance specimens were tested at this time.

Table 4.9-5. RVSP Specimen Orientation Information for Davis-Besse Unit 1

Heat ID	Specimen ID	Orientation
Tensile Specimens		
BCC 241	SS 601-621	Transverse
BCC 241	SS 801-806	Longitudinal
AKJ 233	TT 601-606	Transverse
AKJ 233	TT 801-806	Longitudinal
HAZ -- BCC 241	SS 301-306	Transverse
HAZ -- BCC 241	SS 501-506	Longitudinal
HAZ -- AKJ 233	TT 301-306	Transverse
HAZ -- AKJ 233	TT 501-506	Longitudinal
Weld -- WF 182-1	SS 001-018	Longitudinal
Charpy Specimens		
BCC 241	SS 601-692+	TL
BCC 241	SS 801-820+	LT
AKJ 233	TT 601-637+	TL
AKJ 233	TT 801-820+	LT
HAZ -- BCC 241	SS 301-392+	Transverse, TL
HAZ -- BCC 241	SS 501-516	Transverse, TS
HAZ -- AKJ 233	TT 301-336	Transverse, TL
HAZ -- AKJ 233	TT 501-516	Transverse, TS
Weld -- WF 182-1	SS 001-093	Transverse, TL
HSST plate 02	SS 901-930	LT
0.5 TCT Specimens		
Weld -- WF 182-1	SS 001-040	Transverse, TL
1 TCT Specimens		
Weld -- WF 182-1	SS 001-006	Transverse, TL

5. MATERIALS CROSS-REFERENCE

Since the early RVSPs were required to contain only representative reactor vessel materials, rather than actual reactor vessel materials, some of the B&W plant-specific RVSPs do not contain the materials that would be required by current regulation. Therefore, these programs do not provide materials data which are of direct use for their reactor vessels. However, other RVSPs may contain the material desired and, under the integrated program concept, these data can be applied to the reactor vessel in question.

B&W plant-specific RVSPs contain only two of the four or more different base metals which make up the reactor vessel. In every case, the surveillance base materials are from the monitored vessel. Data for a particular vessel material can be provided from surveillance specimens from the same piece of material or from specimens of a different piece, but same heat, of material (in many cases one heat of steel was sufficient to make several separate plates). When there is no surveillance material for a particular vessel material, data can still be provided by combining the data from specimens of the same material specification: A 302B and A 533B, Class 1 as one group, A 508 Class 2 as another group. Not all surveillance specimens can be related to vessel material, even if the specimens are from the same piece as the vessel metal; if the heat treatments or processing stages of the specimens are significantly different from those of the vessel material, the two may not have similar mechanical properties. This situation should be considered in any analysis of the surveillance data.

The HAZ material situation is very similar to that of the base metal, with the added concern of welding parameters and processing to be considered.

The RVSPs contain only one of the three or more different weld metals that are in the reactor vessel. In every case, the surveillance weldment was made using the same welding process, similar parameters, and the same classes of weld wire (except for the atypical weld)* and flux as the vessel weldments. However, the weld wire heat and/or flux lot differed in most cases. Data for the vessel weld in question can be provided from surveillance specimens made from the same weld wire heat/flux combination (as required by the current revision of 10 CFR 50, Appendix G), or from specimens made from the same wire heat but different flux lot, since the major variable in weld makeup is the filler wire used. When a vessel weld has no surveillance material, data can be provided by combining the data from all weld metal specimens, since the same class of filler wire and flux, and the same welding process and equipment were used in all cases. However, as with the base metal, note that stress-relief time and temperature can have a large effect on mechanical properties, so welds made by the same process and of the same wire heat and flux lot do not necessarily behave the same way. This should be considered in any analysis of the surveillance data.

The cross-reference of reactor vessel beltline materials and surveillance materials is shown on Tables 5-1 through 5-8. These tables key on the vessel material and show the source of data, type of material, type of specimen, and quantity of specimens. For base metal and HAZ material, specimens from the same piece and/or heat of vessel material are listed. For weld metal, specimens from the same wire heat and flux combination and/or wire heat are listed. There are actually more weld metal specimens than those listed, since B&W-manufactured weld metals have been used in a number of irradiated materials evaluation programs, B&W Owners' Group research capsules, and in non-B&W Owners' RVSPs. None of these sources are mentioned here; they will be addressed in a separate B&W report to be prepared presently. The surveillance specimens in Tables 5-1 through 5-8 are listed only by source similarity and do not consider heat treatments and processing.

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*See the appendix.

Table 5.0-1. Surveillance Specimens Applicable to Oconee Unit 1 Reactor Vessel Material

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens: Tensile	CVN	CT
AHR 54	--										
C2197-2	--										
C3265-1	C3265-1	OC-1 -- Baseline	X	X					12	42	
	C3265-1	OCI-A	X	X					4	12	
	C3265-1	OCI-C	X	X					4	12	
	C3265-1	OCI-E	X	X					4	12	
	HAZ -- C3265-1	OC-1 -- Baseline	X	X					12	41	
	HAZ -- C3265-1	OCI-A	X	X						8	
	HAZ -- C3265-1	OCI-C	X	X						8	
	HAZ -- C3265-1	OCI-E	X	X						8	
C3278-1	--										
C2800-1	C2800-2	See C2800-2 vessel material		X							
C2800-2	C2800-2	OC-1 -- Baseline	X	X					12	39	
	C2800-2	OCI-B	X	X					4	18	
	C2800-2	OCI-D	X	X					4	18	
	C2800-2	OCI-E	X	X					4	18	
	HAZ -- C2800-2	OC-1 -- Baseline	X	X					12	41	
	HAZ -- C2800-2	OCI-B	X	X					4	10	
	HAZ -- C2800-2	OCI-D	X	X					4	10	
	HAZ -- C2800-2	OCI-E	X	X					4	10	
122S347VA1	--										
SA 1135	--										
SA 1229	--										
WF 25	WF 25	TMI-1 -- Baseline			X			X	6	18	
	WF 25	TMI 1-A			X			X	4	8	
	WF 25	TMI 1-C			X			X	4	8	
	WF 25	TMI 1-E			X			X	4	8	
SA 1585	--										
WF 9	--										
SA 1073	--										
SA 1493	--										
SA 1430	--										

Table 5.0-2. Surveillance Specimens Applicable to Oconee Unit 2 Reactor Vessel Material

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens: Tensile	CVN	CT
AMX 77	--										
AAW 163	AAW 163	OC-2 -- Baseline	X	X					12	54	
	AAW 163	OCII-A	X	X					4	12	
	AAW 163	OCII-C	X	X					4	12	
	AAW 163	OCII-E	X	X					4	12	
	HAZ -- AAW 163	OC-2 -- Baseline	X	X					12	54	
	HAZ -- AAW 163	OCII-A	X	X						8	
	HAZ -- AAW 163	OCII-C	X	X						8	
	HAZ -- AAW 163	OCII-E	X	X						8	
AWG 164	AWG 164	OC-2 -- Baseline	X	X					12	53	
	AWG 164	OCII-B	X	X					4	18	
	AWG 164	OCII-D	X	X					4	18	
	AWG 164	OCII-F	X	X					4	18	
	HAZ -- AWG 164	OC-2 -- Baseline	X	X					12	53	
	HAZ -- AWG 164	OCII-B	X	X					4	10	
	HAZ -- AWG 164	OCII-D	X	X					4	10	
	HAZ -- AWG 164	OCII-F	X	X					4	10	
122T293VA1	--										
WF 154	WF 112	OC-1 -- Baseline				X		X	6	21	
	WF 112	OCI-A				X		X	4	8	
	WF 112	OCI-C				X		X	4	8	
	WF 112	OCI-E				X		X	4	8	
	WF 193	ANO-1 -- Baseline				X		X	6	27	
	WF 193	AN1-A				X		X	4	8	
	WF 193	AN1-C				X		X	4	8	
	WF 193	AN1-E				X		X	4	8	
	WF 193	RS-1 -- Baseline				X		X	5	14	8
	WF 193	RS1-A				X		X	2	12	
	WF 193	RS1-B				X		X	2	12	8
	WF 193	RS1-C				X		X	2	12	
	WF 193	RS1-D				X		X	2	12	8
	WF 193	RS1-E				X		X	2	12	
	WF 193	RS1-F				X		X	2	12	8
WF 25	WF 25	TMI-1 -- Baseline			X			X	6	18	
	WF 25	TMI1-A			X			X	4	8	
	WF 25	TMI1-C			X			X	4	8	
	WF 25	TMI1-E			X			X	4	8	
WF 112	WF 112	See WF 154 vessel material			X			X			
	WF 193	See WF 154 vessel material				X		X			

Table 5.0-3. Surveillance Specimens Applicable to Three Mile Island Unit 1 Reactor Vessel Material

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wi	Vessel weld	Surveillanc weld	Type/number of test specimens		
									Tensile	CVN	CT
ARY 59	--										
C2789-1	C2789-1	See C2789-2 vessel material		X							
C2789-2	C2789-2	TMI-1 -- Baseline	X	X					12	56	
	C2789-2	TMI1-A	X	X					4	12	
	C2789-2	TMI1-C	X	X					4	12	
	C2789-2	TMI1-E	X	X					4	12	
	HAZ -- C2789-2	TMI-1 -- Baseline	X	X					12	59	
	HAZ -- C2789-2	TMI1-A	X	X						8	
	HAZ -- C2789-2	TMI1-C	X	X						8	
	HAZ -- C2789-2	TMI1-E	X	X						8	
C3307-1	C3307-1	TMI-1 -- Baseline	X	X					12	54	
	C3307-1	TMI1-B	X	X					4	18	
	C3307-1	TMI1-D	X	X					4	18	
	C3307-1	TMI1-E	X	X					4	18	
	HAZ -- C3307-1	TMI-1 -- Baseline	X	X					12	51	
	HAZ -- C3307-1	TMI1-B	X	X					4	10	
	HAZ -- C3307-1	TMI1-D	X	X					4	10	
	HAZ -- C3307-1	TMI1-E	X	X					4	10	
C3251-1	--										
122T229VA1	--										
WF 70	WF 209-1	OC-2 -- Baseline				X		X	6	15	
	WF 209-1	OCII-A				X		X	4	8	
	WF 209-1	OCII-C				X		X	4	8	
	WF 209-1	OCII-E				X		X	4	8	
	WF 209-1	OC-3 -- Baseline				X		X	6	21	
	WF 209-1	OCIII-A				X		X	2	12	
	WF 209-1	OCIII-B				X		X	2	12	
	WF 209-1	OCIII-C				X		X	2	12	
	WF 209-1	OCIII-D				X		X	2	12	
	WF 209-1	OCIII-E				X		X	2	12	
	WF 209-1	OCIII-F				X		X	2	12	
	WF 209-1	CR-3 -- Baseline				X		X			14-15
	WF 209-1	CR3-B				X		X			3-5
	WF 209-1	CR3-D				X		X			4
	WF 209-1	CR3-F				X		X			5-6
WF 25	WF 25	TMI-1 -- Baseline			X			X	6	18	
	WF 25	TMI1-A			X			X	4	8	
	WF 25	TMI1-C			X			X	4	8	
	WF 25	TMI1-E			X			X	4	8	
WF 67	--										
WF 8	--										
SA 1526	WF 25	See WF 25 vessel material				X		X			
SA 1494	--										

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Table 5.0-4. Surveillance Specimens Applicable to Crystal River Unit 3 Reactor Vessel Material

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens: Tensile	CVN	CT
AZJ 94	--										
C4344-1	C4344-1	CR-3 -- Baseline	X	X					12	30	
	C4344-1	CR3-A	X	X					2	12	
	C4344-1	CR3-B	X	X					2	12	3
	C4344-1	CR3-C	X	X					2	12	
	C4344-1	CR3-D	X	X					2	12	3-4
	C4344-1	CR3-E	X	X					2	12	
	C4344-1	CR3-F	X	X					2	12	1-2
	HAZ -- C4344-1	CR-3 -- Baseline	X	X					12	30	
	HAZ -- C4344-1	CR3-A	X	X						12	
	HAZ -- C4344-1	CR3-B	X	X						12	
	HAZ -- C4344-1	CR3-C	X	X						12	
	HAZ -- C4344-1	CR3-D	X	X						12	
	HAZ -- C4344-1	CR3-E	X	X						12	
	HAZ -- C4344-1	CR3-F	X	X						12	
	C4344-2	See C4344-2 vessel material		X							
C4344-2	C4344-2	CR-3 -- Baseline							12	30	
	C4344-2	CR3-A								6	
	C4344-2	CR3-C								6	
	C4344-2	CR3-E								6	
	HAZ -- C4344-2	CR-3 -- Baseline							12	30	
	HAZ -- C4344-2	CR3-A								6	
	HAZ -- C4344-2	CR3-C								6	
	HAZ -- C4344-2	CR3-E								6	
	C4344-1	See C4344-1 vessel material									
C4347-1	--										
C4347-2	--										
124W295VA1	--										
SA 1769	--										
WF 169-1	--										
WF 8	--										
WF 18	--										
WF 70	WF 209-1	OC-2 -- Baseline				X		X	6	15	
	WF 209-1	OCII-A				X		X	4	8	
	WF 209-1	OCII-C				X		X	4	8	
	WF 209-1	OCII-E				X		X	4	8	
	WF 209-1	OC-3 -- Baseline				X		X	6	21	
	WF 209-1	OCIII-A				X		X	2	12	
	WF 209-1	OCIII-B				X		X	2	12	
	WF 209-1	OCIII-C				X		X	2	12	
	WF 209-1	OCIII-D				X		X	2	12	
	WF 209-1	OCIII-E				X		X	2	12	
	WF 209-1	OCIII-F				X		X	2	12	

Table 5.0-4. (Cont'd)

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens: Tensile	CVN	CT
	WF 209-1	CR-3 -- Baseline				X		X			14-15
	WF 209-1	CR3-B				X		X			3-5
	WF 209-1	CR3-D				X		X			4
	WF 209-1	CR3-F				X		X			5-6
SA 1580	--										
WF 154	WF 193	ANO-1 -- Baseline				X		X	6	27	
	WF 193	AN1-A				X		X	4	8	
	WF 193	AN1-C				X		X	4	8	
	WF 193	AN1-E				X		X	4	8	
	WF 193	RS-1 -- Baseline				X		X	5	14	8
	WF 193	RS1-A				X		X	2	12	
	WF 193	RS1-B				X		X	2	12	8
	WF 193	RS1-C				X		X	2	12	
	WF 193	RS1-D				X		X	2	12	8
	WF 193	RS1-E				X		X	2	12	
	WF 193	RS1-F				X		X	2	12	8
	WF 112	OC-1 -- Baseline				X		X	6	21	
	WF 112	OCI-A				X		X	4	8	
	WF 112	OCI-C				X		X	4	8	
	WF 112	OCI-E				X		X	4	8	

Table 5.0-5. Surveillance Specimens Applicable to Arkansas Nuclear One Unit 1 Reactor Vessel Material

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens		
									Tensile	CVN	CT
AYN 131	--										
C5120-2	--										
C5114-2	C5114-2	ANO-1 -- Baseline	X	X					12	54	
	C5114-2	AN1-B	X	X					4	18	
	C5114-2	AN1-D	X	X					4	18	
	C5114-2	AN1-F	X	X					4	10	
	HAZ -- C5114-2	ANO-1 -- Baseline	X	X					12	53	
	HAZ -- C5114-2	AN1-B	X	X					4	10	
	HAZ -- C5114-2	AN1-D	X	X					4	10	
	HAZ -- C5114-2	AN1-F	X	X					4	10	
	C5114-1	See C5114-1 base material		X							
C5120-1	--										
C5114-1	C5114-1	ANO-1 -- Baseline	X	X					12	53	
	C5114-1	AN1-A	X	X					4	12	
	C5114-1	AN1-C	X	X					4	12	
	C5114-1	AN1-E	X	X					4	12	
	HAZ -- C5114-1	ANO-1 -- Baseline	X	X					11	53	
	HAZ -- C5114-1	AN1-A	X	X						8	
	HAZ -- C5114-1	AN1-C	X	X						8	
	HAZ -- C5114-1	AN1-E	X	X						8	
	C5114-2	See C5114-2 base material		X							
125W609VA1	--										
WF 182-1	WF 182-1	DB-1 -- Baseline			X			X	5	19	12
	WF 182-1	TE1-A			X			X	2	12	
	WF 182-1	TE1-B			X			X	2	12	8
	WF 182-1	TE1-C			X			X	2	12	
	WF 182-1	TE1-D			X			X	2	12	8
	WF 182-1	TE1-E			X			X	2	12	
	WF 182-1	TE1-F			X			X	2	12	8
WF 112	WF 112	OC-1 -- Baseline			X			X	6	21	
	WF 112	OCI-A			X			X	4	8	
	WF 112	OCI-C			X			X	4	8	
	WF 112	OCI-E			X			X	4	8	
	WF 193	ANO-1 -- Baseline				X		X	6	27	
	WF 193	AN1-A				X		X	4	8	
	WF 193	AN1-C				X		X	4	8	
	WF 193	AN1-E				X		X	4	8	
	WF 193	RS-1 -- Baseline				X		X	5	14	8
	WF 193	RS1-A				X		X	2	12	
	WF 193	RS1-B				X		X	2	12	8
	WF 193	RS1-C				X		X	2	12	
	WF 193	RS1-D				X		X	2	12	8
	WF 193	RS1-E				X		X	2	12	
	WF 193	RS1-F				X		X	2	12	8
SA 1788	--										
WF 18	--										

Table 5.0-6. Surveillance Specimens Applicable to Oconee Unit 3 Reactor Vessel Material

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens: Tensile	CVN	CT
4680	--										
AWS 192	AWS 192	OC-3 -- Baseline	X	X					12	56	
	AWS 192	OCIII-A	X	X						9	
	AWS 192	OCIII-B	X	X						6	
	AWS 192	OCIII-C	X	X						9	
	AWS 192	OCIII-D	X	X						6	
	AWS 192	OCIII-E	X	X						9	
	AWS 192	OCIII-F	X	X						6	
	HAZ -- AWS 192	OC-3 -- Baseline	X	X					12	56	
	HAZ -- AWS 192	OCIII-B	X	X						6	
	HAZ -- AWS 192	OCIII-D	X	X						6	
	HAZ -- AWS 192	OCIII-F	X	X						6	
ANK 191	ANK 191	OC-3 -- Baseline	X	X					12	59	
	ANK 191	OCIII-A	X	X					2	21	
	ANK 191	OCIII-B	X	X					2	12	
	ANK 191	OCIII-C	X	X					2	21	
	ANK 191	OCIII-D	X	X					2	12	
	ANK 191	OCIII-E	X	X					2	21	
	ANK 191	OCIII-F	X	X					2	12	
	HAZ -- ANK 191	OC-3 -- Baseline	X	X					12	48	
	HAZ -- ANK 191	OCIII-A	X	X						12	
	HAZ -- ANK 191	OCIII-B	X	X						12	
	HAZ -- ANK 191	OCIII-C	X	X						12	
	HAZ -- ANK 191	OCIII-D	X	X						12	
	HAZ -- ANK 191	OCIII-E	X	X						12	
	HAZ -- ANK 191	OCIII-F	X	X						12	
417543-1	--										
WF 200	WF 182-1	DB-1 -- Baseline				X		X	5	19	12
	WF 182-1	TE1-A				X		X	2	12	
	WF 182-1	TE1-B				X		X	2	12	8
	WF 182-1	TE1-C				X		X	2	12	
	WF 182-1	TE1-D				X		X	2	12	8
	WF 182-1	TE1-E				X		X	2	12	
	WF 182-1	TE1-F				X		X	2	12	8
WF 67	--										
WF 70	WF 209-1	OC-2 -- Baseline				X		X	6	15	
	WF 209-1	OCII-A				X		X	4	8	
	WF 209-1	OCII-C				X		X	4	8	
	WF 209-1	OCII-E				X		X	4	8	
	WF 209-1	OC-3 -- Baseline				X		X	6	21	
	WF 209-1	OCIII-A				X		X	2	12	
	WF 209-1	OCIII-B				X		X	2	12	
	WF 209-1	OCIII-C				X		X	2	12	
	WF 209-1	OCIII-D				X		X	2	12	
	WF 209-1	OCIII-E				X		X	2	12	
	WF 209-1	OCIII-F				X		X	2	12	
	WF 209-1	CR-3 -- Baseline				X		X			14-15
	WF 209-1	CR3-B				X		X			3-5
	WF 209-1	CR3-D				X		X			4
	WF 209-1	CR3-F				X		X			5-6
WF 169-1	--										

Table 5.0-7. Surveillance Specimens Applicable to Rancho Seco Unit 1 Reactor Vessel Material

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens Tensile	CVN	CT
FV 4823	--										
C5062-1	C5062-1	RS-1 -- Baseline	X	X					12	29	
	C5062-1	RS1-A	X	X					2	12	
	C5062-1	RS1-B	X	X					2	12	
	C5062-1	RS1-C	X	X					2	12	
	C5062-1	RS1-D	X	X					2	12	
	C5062-1	RS1-E	X	X					2	12	
	C5062-1	RS1-F	X	X					2	12	
	HAZ -- C5062-1	RS-1 --Baseline	X	X					12	30	
	HAZ -- C5062-1	RS1-A	X	X						12	
	HAZ -- C5062-1	RS1-B	X	X						12	
	HAZ -- C5062-1	RS1-C	X	X						12	
	HAZ -- C5062-1	RS1-D	X	X						12	
	HAZ -- C5062-1	RS1-E	X	X						12	
	HAZ -- C5062-1	RS1-F	X	X						12	
C5062-2	C5062-1	See C5062-1 vessel material		X							
C5070-1	C5070-1	RS-1 -- Baseline	X	X					12	30	
	C5070-1	RS1-A	X	X						6	
	C5070-1	RS1-C	X	X						6	
	C5070-1	RS1-E	X	X						6	
	HAZ -- C5070-1	RS-1 -- Baseline	X	X					12	30	
	HAZ -- C5070-1	RS1-A	X	X						6	
	HAZ -- C5070-1	RS1-C	X	X						6	
	HAZ -- C5070-1	RS1-E	X	X						6	
C5070-2	C5070-1	See C5070-1 vessel material		X							
WF 233	--										
WF 154	WF 112	OC-1 -- Baseline				X		X	6	21	
	WF 112	OC1-A				X		X	4	8	
	WF 112	OC1-C				X		X	4	8	
	WF 112	OC1-E				X		X	4	8	
	WF 193	ANO-1 -- Baseline				X		X	6	27	
	WF 193	AN1-A				X		X	4	8	
	WF 193	AN1-C				X		X	4	8	
	WF 193	AN1-E				X		X	4	8	
	WF 193	RS-1 -- Baseline				X		X	5	14	8
	WF 193	RS1-A				X		X	2	12	
	WF 193	RS1-B				X		X	2	12	8
	WF 193	RS1-C				X		X	2	12	
	WF 193	RS1-D				X		X	2	12	8
	WF 193	RS1-E				X		X	2	12	
	WF 193	RS1-F				X		X	2	12	8
WF 29	--										

Table 5.0-7. (Cont'd)

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens: Tensile	CVN	CT
WF 70	WF 209-1	OC-2 -- Baseline				X		X	6	15	
	WF 209-1	OCII-A				X		X	4	8	
	WF 209-1	OCII-C				X		X	4	8	
	WF 209-1	OCII-E				X		X	4	8	
	WF 209-1	OC-3 -- Baseline				X		X	6	21	
	WF 209-1	OCIII-A				X		X	2	12	
	WF 209-1	OCIII-B				X		X	2	12	
	WF 209-1	OCIII-C				X		X	2	12	
	WF 209-1	OCIII-D				X		X	2	12	
	WF 209-1	OCIII-E				X		X	2	12	
	WF 209-1	OCIII-F				X		X	2	12	
	WF 209-1	CR-3 -- Baseline				X		X			14-15
	WF 209-1	CR3-B				X		X			3-5
	WF 209-1	CR3-D				X		X			4
	WF 209-1	CR3-F				X		X			5-6

Table 5.0-8. Surveillance Specimens Applicable to Davis-Besse Unit 1 Reactor Vessel Material

Vessel material	Surveillance material	Source	Same piece as vessel material	Same heat as vessel material	Same wire and flux	Same wire	Vessel weld	Surveillance weld	Type/number of test specimens: Tensile	CVN	CT
ADB 203	--										
AKJ 233	AK 233	DB-1 -- Baseline	X	X					12	30	
	AK 233	TE1-A	X	X						6	
	AK 233	TE1-C	X	X						6	
	AK 233	TE1-E	X	X						6	
	HA -- AKJ 233	DB-1 -- Baseline	X	X					12	30	
	HA -- AKJ 233	TE1-A	X	X						6	
	HA -- AKJ 233	TE1-C	X	X						6	
	HA -- AKJ 233	TE1-E	X	X						6	
BCC 241	BC 241	DB-1 -- Baseline	X	X					12	39	
	BC 241	TE1-A	X	X					2	12	
	BC 241	TE1-B	X	X					2	12	
	BC 241	TE1-C	X	X					2	12	
	BC 241	TE1-D	X	X					2	12	
	BC 241	TE1-E	X	X					2	12	
	BC 241	TE1-F	X	X					2	12	
	HAZ -- BCC 241	DB-1 -- Baseline	X	X					12	36	
	HAZ -- BCC 241	TE1-A	X	X						12	
	HAZ -- BCC 241	TE1-B	X	X						12	
	HAZ -- BCC 241	TE1-C	X	X						12	
	HAZ -- BCC 241	TE1-D	X	X						12	
	HAZ -- BCC 241	TE1-E	X	X						12	
	HAZ -- BCC 241	TE1-F	X	X						12	
122Y384VA1	--										
WF 232	--										
WF 233	--										
WF 182-1	WF 182-1	DB-1 -- Baseline			X			X	5	19	12
	WF 182-1	TE1-A			X			X	2	12	
	WF 182-1	TE1-B			X			X	2	12	8
	WF 182-1	TE1-C			X			X	2	12	
	WF 182-1	TE1-D			X			X	2	12	8
	WF 182-1	TE1-E			X			X	2	12	
	WF 182-1	TE1-F			X			X	2	12	8

6. CERTIFICATION

This report is an accurate description, so far as can be determined, of the materials in the B&W Owners' Group reactor vessels and reactor vessel material surveillance programs.

John D. Aadland 12/12/84

J. D. Aadland Date
Materials and Chemical Engineering

This report has been reviewed and is an accurate description, so far as can be determined, of the materials in the B&W Owners' Group reactor vessels and reactor vessel material surveillance programs.

A. L. Lowe, Jr., P.E. 15 Feb 1985

A. L. Lowe, Jr., P.E. Date
Project Technical Manager

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Appendix

GLOSSARY

Arkansas Nuclear One, Unit 1 (ANO-1)	Arkansas Power & Light's nuclear power station located in Russellville, Arkansas.
Automatic Submerged-Metal Arc (ASA)	A welding process used in the manufacture of B&W-made reactor vessels.
Atypical weld	Weldment originally designated as WF 209 that is now known to be off-chemistry due to an incorrectly identified coil of weld wire. Subject of report BAW-10144A.
Axial	Forging test specimen orientation defined in ASTM Standard A 370.
Baseline	Test specimens or data used to define unirradiated mechanical properties of reactor vessel materials. Sometimes erroneously used to describe base metal contained in surveillance capsules.
Base metal/base material	The wrought product (forging or rolled plate) segment of a reactor vessel or surveillance program.
Beltline	That section of a reactor vessel which undergoes sufficient radiation damage to be considered as a potential limiting material of construction. Defined in 10 CFR 50, Appendix G.
Capsule	A small closed cylinder used to house test specimens, dosimeters, and thermal monitors in a protective environment to simulate the irradiated reactor vessel condition.
Core region	The cylindrical region of the reactor vessel in the vicinity directly opposite the core, composed of the upper shell, lower shell, and, in the case of Oconee 1 only, the intermediate shell.
Correlation material	Also referred to as correlation monitor material, reference material, or HSST Plate 02. This is a plate of SA 533 Grade B Class 1 that has been approved by ASTM Committee E-10 as standard reference material for surveillance monitoring. Test specimens drawn from this plate are included in some surveillance capsules.

Crystal River Unit 3 (CR-3)	Florida Power Company's nuclear power station located in Red Level, Florida.
Compact Fracture Toughness (CT)	Also referred to as compact tension, compact toughness, and compact fracture, specimen used in J-Integral and K_{IC} testing. Also, Round Compact Fracture Toughness (RCT), referring to the shape of the specimen. When seen as 1TCT, for example, the first T stands for thickness and the number preceding it refers to the specimen's thickness in inches.
Charpy V-Notch (CVN)	Designation for the Charpy V-Notch impact test or specimen. Also referred to as Charpy impact or simply Charpy. Sometimes denoted C_v.
Davis-Besse Unit 1 (DB-1)	Toledo Edison Company's nuclear power station located at Oak Harbor, Ohio.
Dutchman	A full circumferential ring forging toward the bottom of a reactor vessel, between the lower shell and lower head, for transition from shell geometry to head geometry.
Fluence	Term used to describe accumulated radiation exposure of a reactor vessel location or surveillance capsule. Units of interest are usually n/cm^2 (E > 1 MeV), although n/cm^2 (E > 0.1 MeV) and displacements per atom (DPA) are also used.
Flux	1. Granular material used to cover molten metal during the welding process to prevent the molten metal from reacting with air and to improve weld quality. 2. Radiation exposure rate, usually expressed as n/cm^2-s.
Forging	A full circumferential portion of the reactor vessel fabricated by a forging, as opposed to the hot rolling process.
Guest plant	A nuclear plant for which surveillance capsules are irradiated at a different plant's reactor vessel.
Heat-Affected Zone (HAZ)	A region in base metal immediately adjacent to a weld and affected by the welding heat. The term applies to both reactor vessel and surveillance specimens.

Host plant	A nuclear plant which serves as the irradiation site for another plant's surveillance capsules.
Heavy Section Steel Technology (HSST)	Title of a program conducted by Oak Ridge National Laboratory in Oak Ridge, Tennessee.
Integrated Reactor Vessel Materials Surveillance Program (IRVSP)	A program developed by the B&W Owners' Group to share pertinent data and allow the irradiation of capsules from plants without capsule holder tubes. Also referred to as Integrated Program.
Intermediate shell	The top segment of the core region, applicable to Oconee 1 only.
Lower nozzle belt	A full circumferential ring forging above the core barrel, in which the vessel inlet and outlet nozzles are partially situated. Not applicable to Davis-Besse 1, which has a single nozzle belt forging.
Lower shell	The bottom section of the core region, located between the upper shell and the dutchman. It may be either a single forging or two semi-cylindrical rolled plates welded together.
Nozzle belt	An area between the core region and the flange, in which the vessel inlet and outlet nozzles are situated. The nozzle belt is made up of a single full circumferential forging in Davis-Besse 1 only; in all other vessels there are two forgings (upper and lower) composing the nozzle belt.
Nozzle belt dropout (NBD)	A solid cylindrical section removed from the nozzle belt forging to allow for the installation of a nozzle. The dropouts, which may contain weld metal, are a source of test specimens for the B&W Owners' Group research capsules.
Nuclear Steam System (NSS)	That part of the nuclear power plant consisting of the reactor vessel, steam generators, coolant pumps, and associated piping, systems, and instrumentation supplied by B&W. This term is often used to refer to a plant, as in NSS-3 for Oconee 1.

Oconee 1, Oconee 2, Oconee 3	Shorthand for Duke Power Company's three nuclear power stations located in Seneca, South Carolina.
Orientation	The directional relationship a test specimen has with its source material, particularly with the material's principal working direction. An orientation convention for B&W-supplied surveillance materials is described in this report.
Plant-specific	Term used for something pertaining to one plant only. A plant-specific surveillance program denotes test specimens irradiated in one plant for that plant.
Plate	A component of the reactor vessel produced by a hot rolling fabrication process. Most B&W-supplied reactor vessels have core regions fabricated from plate material.
Rancho Seco Unit 1 (RS-1)	Sacramento Municipal Utility District's nuclear power station in Clay Station, California.
Research capsule	Large capsules containing weld metal test specimens which are being irradiated and tested within the B&W Owners' Group Integrated Reactor Vessel Materials Surveillance Program.
Reactor Vessel Materials Surveillance Program (RVSP)	A plant-specific program designed to monitor one vessel only.
Surveillance Capsule Holder Tube (SCHT)	A device which holds two capsules in place within the reactor vessel. Originally located on the inside surface of early reactor vessels; due to damage, they were installed on the outside of the thermal shields in host plants.
Surveillance	Term used to describe the monitoring of radiation damage to the reactor vessel by the accelerated irradiation of vessel material test specimens.
Surveillance weld	A weldment made specifically to provide test specimens for a surveillance program. The weld is made according to the vessel welding procedure, using excess vessel material as base metal.

Tangential	Forging test specimen orientation, defined in ASTM Standard A 370.
Tensile	Term used for specimen to be used in a tension test to measure strength and ductility. Also referred to as a tension test specimen.
Three Mile Island Unit 1 (TMI-1)	General Public Utilities' (Nuclear) nuclear power station located in Londonderry Township, Pennsylvania.
Upper nozzle belt	A full circumferential ring forging above the lower nozzle belt, in which the vessel inlet and outlet nozzles are partially situated. Not applicable to Davis-Besse 1, which has a single nozzle belt forging.
Upper shell	The upper (Oconee 1 only: middle) section of the core region lcoated between the lower nozzle belt forging (Davis-Besse 1 only: nozzle belt forging) and the lower shell. It may be either a single forging or two semi-cylindrical rolled plates welded together.
Weld metal	The material deposited during a welding process, making it a cast material. Made up mostly of welded filler wire, with some dilution due to the melting of base metal.
Weld wire/filler wire	Steel wire melted during the welding process to join base metal sections. Weld wire heats are groupings of wire coils that were manufactured from the same heat of steel.

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