5.3 REACTOR VESSEL

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Material Specifications

→(EC-1020, R307)
The principal ferritic materials used in the reactor vessel are listed in Table 5.2-3. These materials were specified to be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition including Summer 1971 Addenda, except for the Replacement Reactor Vessel Closure Head materials which were specified to be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition through 2000 Addenda.
←(EC-1020, R307)

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The reactor vessel is a right circular cylinder with two hemispherical heads. No special manufacturing methods that could compromise the integrity of the vessel are used. The lower head is permanently welded to the lower end of the reactor vessel shell but the upper closure head can be removed to provide access to the reactor vessel internals. The head flange is drilled to match the vessel flange stud bolt locations. The stud bolts are fitted with spherical washers located between the closure nuts and the head flange. These washers maintain stud alignment during boltup when flexing of the head must be accommodated. The lower surface of the head flange is machined to provide a mating surface for the vessel closure seals.

The vessel flange is a forged ring with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seals. An externally tapered transition section connects the flange to the cylindrical shell.

Sealing is accomplished by using two silver-plated, NiCrFe alloy, self-energized 0-rings.

Nozzles are provided in the closure head for nuclear instrumentation and control element drive mechanisms (CEDM).

The inlet and outlet nozzles are located radially on a common plane just below the vessel flange. Ample thickness in this vessel course provides most of the reinforcement required for the nozzles. Additional reinforcement is provided for the individual nozzle attachments. A boss located around the outlet nozzles on the inside diameter of the vessel wall provides a mating surface for the core support barrel and guides the outlet coolant flow. This boss and the outlet sleeve on the core support barrel are machined to a common contour to minimize reactor coolant bypass leakage. Shell sections are joined to the nozzle region by a transition section.

Snubbers built into the lower portion of the reactor vessel shell limit the amplitude of flow-induced vibrations in the core support barrel.

5.3.1.3 Special Methods for Nondestructive Examination

Prior to, during, and after fabrication of the reactor vessel, nondestructive tests based upon Section III of the ASME Boiler and Pressure Vessel Code were performed on all welds, forgings, and plates as indicated.

All full-penetration, pressure-containing welds were 100 percent radiographed to the standards of Section III of the ASME Boiler and Pressure Vessel Code. Weld preparation areas, back-chip areas, and final weld surfaces were magnetic-particle or dye-penetrant examined. Other pressure-containing welds, such as used for the attachments of nonferrous nickel-chromium-iron mechanism housings, vents, and instrument housings to the reactor vessel head, were inspected by liquid-penetrant tests of the root pass, the lesser of one- third of the thickness or each 1/2 in. of weld deposit, and the final surface. Additionally, the base metal weld preparation area was magnetic-particle examined prior to overlay with nickel-chromium-iron weld metal.

All forgings were inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings were tested using shear wave techniques. Rejection under longitudinal beam inspection, with calibration so that the first back reflection is at least 75 percent of screen height, was based on indications causing complete loss of back reflection (when not associated with geometrical configuration).

All carbon-steel forgings and ferrite welds are also subjected to magnetic-particle examination after stress relief. Rejection is based on relevent indication of:

a) Any cracks and linear indications

b) Rounded indications with dimensions greater than 3/16 in.

c) Four or more rounded indications in a line separated by less than 1/16 in. edge to edge

d) Ten or more rounded indications in any 6.0 in.^{2} in the most unfavorable locations

Plates were subjected to ultrasonic examination using straight beam techniques. Rejection was based on areas producing a continuous total loss of back reflection with a frequency and instrument adjustment that produce a minimum of 50 to a maximum of 75 percent of full scale reference back reflection from the opposite side of a sound area of the plate.

→(DRN 06-872, R15)

Any defect that showed a total loss of back reflection that could not be contained within a circle whose diameter is the greater of three inches or one-half the plate thickness was unacceptable. Two or more defects smaller than described above, which cause a complete loss of back reflection, shall be unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect, unless the defects are contained within the area described above. All carbon and low-alloy steel products were magnetic-particle examined after accelerated cooling to the magnetic-particle acceptance standard cited above.

←(DRN 06-872, R15)

→(DRN 06-872, R15)
Nondestructive testing of a vessel was performed throughout fabrication. The nondestructive examination requirements including calibration methods, instrumentation, sensitivity, and reproducibility of data, are in accordance with requirements of the ASME B&PV Code, Section III. (See Table 5.2-1). Strict quality control was maintained in critical areas such as calibration of test instruments.
←(DRN 06-872, R15)

All vessel bolting material received ultrasonic and magnetic-particle examination during the manufacturing process.

The bolting material receives a straight-beam, radial-scan, ultrasonic examination with a search unit not exceeding one square in. area. The standard for rejection was 50 percent loss of first back reflection or an indication in excess of 20 percent of the height of the back reflection. All hollow material receives a second ultrasonic examination using angle beam, radial scan with a search unit not exceeding one square in. in area. A reference specimen of the same composition and thickness containing a notch (located on the inside surface) one in. in length and a depth of three percent of nominal section thickness, or 3/8 in., whichever is less, was used for calibration.

Any indications exceeding the calibration notch amplitude are unacceptable. Use of these techniques ensures that no materials that have unacceptable flaws, observable cracks, or sharply defined linear defects were used.

→(DRN 06-911, R15)
The magnetic-particle inspection was performed both before and after threading of the studs. Axially aligned defects whose lengths were greater than one in. and nonaxial defects were unacceptable.
←(DRN 06-911, R15)

Upon completion of all postweld heat treatments, the reactor vessel was hydrostatically tested at 3125 psig after which all accessible ferritic weld surfaces, including those of welds used to repair material, were magnetic-particle inspected in accordance with Section III of the ASME Code.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Special controls for ferritic and austenitic stainless steels are as follows:

→(DRN 00-1059, R11-A)
a) Regulatory Guide 1.31, Control of Stainless Steel Welding is addressed in Subsection 5.2.3.4.

b) Regulatory Guide 1.34, Control of Electroslag Weld Properties is addressed in Subsection 5.2.3.3.

c) Regulatory Guide 1.43, Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components is addressed in Subsection 5.2.3.3.

d) Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel is addressed in Subsection 5.2.3.4.
←(DRN 00-1059, R11-A)

→(DRN 00-1059, R11-A)

e) Regulatory Guide 1.50, Control of Preheat Temperature for Welding of Low-Alloy Steel is addressed in Subsection 5.2-3-3.

f) Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility is addressed in Subsection 5.2.3.3.

←(DRN 00-1059, R11-A)

g) Regulatory Guide 1.99, Effects of Residual Element on Predicted Radiation Damage to Reactor Vessel Materials

→(DRN 03-2059, R14)

Westinghouse previously took exception to the methods and procedure for predicting radiation damage to pressure vessel steels contained in Regulatory Guide 1.99. Westinghouse's formal position on Regulatory Guide 1.99 was forwarded to the U.S. NRC in September, 1975 [1]. The methods contained in Regulatory Guide 1.99 for predicting RT_{NDT} shift and Charpy upper shelf energy decreases with irradiation are not appropriate for determining the irradiation behavior of A533-B, Class 1, materials. The methods utilized are based on non-A533-B materials data and incorporate incorrect assumptions concerning the irradiation behavior of vessel materials.

The curve shown in Figure 5.3-1 is utilized for predicting the RT_{NDT} shift of reactor vessel material with low copper content. The curve is based on 550 °F irradiation data for A533-B materials. The data base was collected from published works on the subject of irradiation damage in reactor vessel materials and from test data generated by a joint research program with Westinghouse (then Combustion Engineering), NRC and the Naval Research Laboratory (NRL). Table 5.3-1 lists this data. The indicated literature references for the data are listed in Table 5.3-2. The RT_{NDT} shift prediction curves are shown in relation to the data in Figure 5.3-1. Weld and plate irradiation behavior is considered separately, because research has shown that some weld metal tends to be more sensitive to irradiation damage. The curve is conservatively drawn, envelopes the data, and follows trends described by the data.

Regulatory Guide 1.99 is now used without exception. \

←(DRN 03-2059, R14)

5.3.1.5 Fracture Toughness

→(EC-1020, R307)

The reactor vessel materials were ordered to the ASME 1971 Code Section III, Summer 1971 Addenda, specification, except for the Replacement Reactor Vessel Closure Head materials which were ordered to the ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition through 2000 Addenda. The materials meet the Charpy impact requirements of Subsection NB-2300 (three tests at a temperature to verify 30 ft.-lbs. of absorbed energy). Longitudinal (strong direction) Charpy test data was used to develop RT_{NDT}'s as per Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements". The highest MTEB 5-2 RT_{NDT} value for the Waterford 3 reactor vessel beltline plate material is ±22 °F (lower shell plate M-1004-2). Testing of vessel weld and heat affected zone materials was not required by the applicable code year and addenda and the materials were not available.

←(EC-1020, R307)

Transverse (weak direction) Charpy impact data on plate M-1004-2, weld and heat-affected-zone (HAZ) material is reported in Subsection 5.3.1.6-1, as results of the baseline surveillance testing. This testing, which establishes an RT_{NDT} in a manner consistent with Appendix G 10CFR50, yields an RT_{NDT} for plate M-1004-2 of -20°F. The RT_{NDT} for the weld and HAZ material are shown in Table 5.3-3 and the Charpy data is plotted in Figures 5.3-2, -3 and -4.

The lowest reported Charpy upper shelf energy, longitudinal tests, for the reactor vessel beltline material is 140 ft.-lbs., (intermediate shell plate M-1003-1). Using Branch Technical Position MTEB 5-2 and reducing that value to 65 percent, to reflect the difference between longitudinal and transverse testing, yields a conservative Charpy upper shelf energy of 91 ft.-lbs. This is well in excess of the 75 ft.-lb. requirement of 10CFR50, Appendix G.

5.3.1.6 Material Surveillance

The irradiation surveillance program for Waterford 3 will be conducted to assess the neutron-induced changes in the RT_{NDT} (reference temperature) and the mechanical properties of the reactor vessel materials. Changes in the impact and mechanical properties of the material will be evaluated by the comparison of pre- and post-irradiation test specimens. The capsules containing the surveillance test specimens used for monitoring the neutron induced property changes of the reactor vessel materials will be irradiated under conditions which represent, as closely as practically possible, the irradiation conditions of the reactor vessel.

ASTM E-185-82, Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, and 10CFR50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, present criteria for monitoring changes in the fracture toughness properties of reactor vessel beltline materials through surveillance programs. This reactor vessel surveillance program for Waterford 3 adheres to all of the requirements of ASTM E-185-82 and satisfies 10CFR50, Appendix H.

5.3.1.6.1 Test Materials Selection

→(DRN 03-2059, R14)

Regions of both the intermediate and lower shells of the reactor vessel are nearest to the reactor core and, therefore, sustain the greatest neutron exposure. The material from which surveillance test specimens were manufactured were cut from that plate in the core region which would become the limiting plate with respect to reactor operation during its lifetime. This material (lower shell plate M-1004-2) was selected on the basis of highest initial RT_{NDT} , chemical composition and fluence. The test materials were processed so that they are representative of the materials in the completed reactor vessel. A record of chemical analyses, fabrication history and mechanical properties of the shell plate from which the surveillance test materials were prepared is maintained. The results of mill test chemistries for the six plates of the beltline region of the vessel are presented in Table 5.2-5.

→(DRN 00-1059, R11-A)

←(DRN 03-2059, R14)

Three metallurgically different materials representative of the reactor vessel were used for test specimens. These include base metal, weld metal and heat affected zone (HAZ) materials. In addition to the materials from sections of reactor vessel shell plate, material from a standard heat of ASTM A533-B Class I manganese-molybdenum-nickel steel made available by the USAEC sponsored Heavy Section Steel Technology (HSST) program is also included. This

←(DRN 00-1059, R11-A)

reference material has been fully processed and characterized, and is used for Charpy impact specimen correlation monitors to permit comparisons among the irradiation data from operating power reactors and irradiation data from experimental reactors. Compilation of data generated from post-irradiation tests of the correlation monitors will be carried out by the HSST program.

5.3.1.6.1.1 Base Metal

Base metal test material was manufactured from sections of lower shell plate M-1004-2 which was found to have the combination of RT_{NDT}, chemical composition (Cu and P), and neutron fluence during service, which would first appear to limit the vessel operating lifetime. The unirradiated RT_{NDT}, of each plate in the intermediate and lower shells was determined from drop weight and Charpy data as required in NB-2331 of the ASME Boiler Code, 1971 Edition Summer 1971 Addenda, Section III and is shown in Table 5.2-6. All base metal test material is cut from one shell.

The section of shell plate used for the base metal test material is adjacent to the test material used for ASME Code Section III tests and is at a distance of at least one plate thickness from any water-quenched edge. This material was heat-treated to a metallurgical condition which is representative of the final metallurgical condition of the base metal in the completed reactor vessel.

5.3.1.6.1.2 Welded Plates

Weld metal and HAZ material were produced by welding together sections from the selected base metal plate and another intermediate plate of the reactor vessel. The HAZ test material was manufactured from a section of the same shell plate used for base metal test material.

The sections of shell plate used for weld metal and HAZ test material are adjacent to the test material used for ASME Code Section III tests and are at a distance of at least one plate thickness from any water-quenched edge. The procedure used for making the intermediate-to-lower shell girth weld in the reactor vessel was followed in the manufacture of the weld metal and HAZ test materials. The welded plates were heat-treated to metallurgical conditions that are representative of the final metallurgical conditions of similar materials in the completed reactor vessel.

The test specimens used in establishing the unirradiated RT_{NDT} temperature of the base metal were obtained from 1/4 T (where T is plate thickness) locations of sections of the plate used in the core region. The heat-affected-zone samples were taken from the inner region of the deposited weld metal. The impact properties of the specimen locations are representative of the material through the entire thickness. Use of the RTNDT values obtained from samples taken from the inner regions of the test materials represent a conservative approach for establishing the initial minimum operating temperature and the base for the predicted minimum operating temperature after irradiation, because the advantages of the more favorable RT_{NDT} properties of the surface regions are not taken into consideration.

5.3.1.6.2 Test Specimens

5.3.1.6.2.1 Type and Quantity

→(DRN 00-1059)

The magnitude of the neutron-induced property changes of the reactor vessel materials is determined by comparing the results of tests using irradiated impact and tensile specimens to the results of similar tests using unirradiated specimens. The changes in RT_{NDT} of the vessel materials are determined by adding to the reference temperature (RT_{NDT}) the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 50 ft.-lb. level or that measured at the 35 mils. lateral expansion level, whichever temperature shift is greater. The new values of RT_{NDT} are known as adjusted reference temperature.

←(DRN 00-1059)

Drop weight, Charpy impact, and tensile test specimens were provided for unirradiated tests. Drop weight tests were conducted in accordance with ASTM E-208. Charpy impact tests were conducted in accordance with ASTM E-3. Tensile tests were conducted in accordance with ASTM E-8 and E-21. Correlation of drop weight and Charpy impact tests to establish RT_{NDT} were made in accordance with NB-2300 of the ASME Code, Section III. Charpy impact and tensile test specimens are provided for post-irradiation tests.

The total quantity of specimens furnished for carrying out the overall requirements of this program is presented in Table 5.3-4. A sufficient amount of base metal, weld metal, and HAZ test material to provide two additional sets of test specimens has been obtained with full documentation and identification for future evaluation should the need arise. Each of the test materials has been chemically analyzed for approximately 21 elements, including all those listed in ASTM E-185-82.

5.3.1.6.2.2 Unirradiated Specimens

The type and quantity of test specimens provided for establishing the properties of the unirradiated reactor vessel materials are presented in Table 5.3-5. The data from tests of these specimens provide the basis for determining the neutron-induced property changes of the reactor vessel materials.

a) Drop Weight Test Specimens

Twelve drop weight test specimens of base metal (longitudinal and transverse), weld metal, and HAZ material are provided for establishing the NDTT of the unirradiated surveillance materials. These data form the basis for RT_{NDT} determination. RT_{NDT} is the reference temperature from which subsequent neutron-induced changes are determined.

b) Charpy Impact Test Specimens

Thirty test specimens each of base metal (longitudinal and transverse), weld metal, and HAZ material are provided. This quantity exceeds the minimum number of test specimens recommended by ASTM E-185 for developing a Charpy impact energy transition curve and is intended to provide a sufficient number of data points for establishing accurate Charpy impact energy transition temperatures for these materials. This data, together with the drop weight NDTT, is used to establish an RT_{NDT} for each material.

c) Tensile Test Specimens

Eighteen tensile test specimens each of base metal (longitudinal and transverse), weld metal and HAZ materials are provided. This quantity also exceeds the minimum number of test specimens recommended by ASTM E-185 and is intended to permit a sufficient number of tests for accurately establishing the tensile properties for these materials at a minimum of three test temperatures (e.g., ambient, operating and design).

5.3.1.6.2.3 Irradiated Specimens

Both tensile and impact test specimens are used for determining changes in the static and dynamic properties of the materials due to neutron irradiation. A total of 288 Charpy impact and 54 tensile test specimens is provided. The type and quantity of test specimens provided for establishing the properties of the irradiated materials over the lifetime of the vessel are presented in Table 5.3-6. The attachment of the capsule assemblies to the inside wall of the reactor vessel is described in CENPD-155P. [2]

5.3.1.6.3 Specimen Irradiation

5.3-1.6.3.1 Encapsulation of Specimens

The test specimens are placed within corrosion-resistant capsule assemblies:

a) To prevent corrosion of the carbon steel test specimens by the primary coolant during irradiation.

b) To physically locate the test specimens in selected locations within the reactor

c) To provide a means by which the irradiation conditions (fluence,[a] flux spectrum, temperature) can be determined

d) To facilitate the removal of a desired quantity of test specimens from the reactor when a specified fluence has been attained.

(a) Time integrated neutron flux.

A typical capsule assembly, illustrated in Figure 5.3-5, consists of a series of seven specimen compartments, connected by wedge couplings, and a lock assembly. Each compartment enclosure of the capsule assembly is internally supported by the surveillance specimens and is externally pressure tested to 3125 psia during final fabrication. The wedge couplings also serve as end caps for the specimen compartments and position the compartments within the capsule holders which are attached to the reactor vessel. The lock assemblies fix the locations of the capsules within the holders by exerting axial forces on the wedge coupling assemblies which cause these assemblies to exert horizontal forces against the sides of the holders preventing relative motion. The lock assemblies also serve as a point of attachment for the tooling used to remove the capsules from the reactor.

Each capsule assembly is made up of four Charpy impact test specimen (Charpy impact) compartments and three tensile test specimen - flux/temperature monitor (tensile-monitor) compartments. Each capsule compartment is assigned a unique identification so that a complete record of test specimen location within each compartment can be maintained.

a) Charpy Impact Compartments

Each Charpy impact compartment (Figure 5.3-6) contains 12 impact test specimens. This quantity of specimens provides an adequate number of data points for establishing a Charpy impact energy transition curve for a given irradiated material. Comparison of the unirradiated and irradiated Charpy impact energy transition curves permits determination of the RT_{NDT} changes due to irradiation for the various materials.

The specimens are arranged vertically in four 1 x 3 arrays and are oriented with the notch toward the core. The temperature differential between the specimen and the reactor coolant is minimized by using spacers between the specimens and the compartment and by sealing the entire assembly in an atmosphere of helium.

b) Tensile - Monitor Compartments

Each tensile-monitor compartment (Figure 5.3-7) contains three tensile test specimens, a set of flux spectrum monitors and a set of temperature monitors, for estimating the maximum temperature to which the specimens have been exposed. The entire tensile-monitor compartment is sealed within an atmosphere of helium.

The tensile specimens are placed in a housing machined to fit the compartment. Split spacers are placed around the gage length of the specimen to minimize the temperature differential between the specimen gage length and the coolant.

5.3.1.6.3.2 Flux and Temperature Measurement

The changes in the RT_{NDT} of the reactor vessel materials are derived from specimens irradiated to various fluence levels and in different neutron energy spectra. In order to permit accurate predictions of the RT_{NRT} of the vessel materials, complete information on the neutron flux, neutron energy spectra, and the irradiation temperature of the surveillance specimens must be available.

a) Flux Measurements

Fast neutron flux measurements are obtained by insertion of threshold detectors into each of the six irradiation capsules. Such detectors are particularly suited for the proposed application, because their effective threshold energies lie in the low Mev range. Selection of threshold detectors is based on the recommendations of ASTM E-261, "Method of Measuring Neutron Flux by Radioactive Techniques".

These neutron threshold detectors and the thermal neutron detectors, listed in Table 5.3-7, can be used to monitor the thermal and fast neutron spectra incident on the test specimen. These detectors possess reasonable long half-lives and activation cross sections covering the desired neutron energy range.

One set of flux spectrum monitors is included in each tensile monitor compartment. Each detector is placed inside a sheath which identifies the material and facilitates handling. Cadmium covers are used for those materials (e.g., uranium, nickel, copper and cobalt) which have competing neutron capture activities.

The flux monitors are placed in holes drilled in stainless steel housings as shown in Figure 5.3-7 at three axial locations in each capsule assembly (Figure 5.3-5) to provide an axial profile of the level of fluence which the specimens attain.

In addition to these detectors, the program also includes correlation monitors (Charpy impact test specimens made from a reference heat of ASTM A533-B, Class 1, manganese-molybdenum-nickel steel) which are irradiated along with the specimens made from reactor vessel materials. The changes in impact properties of the reference material provide a cross-check on the dosimetry in any given surveillance program. These changes also provide data for correlating the results from this surveillance program with the results from experimental irradiations and other reactor surveillance programs using specimens of the same reference material.

b) Temperature Estimates

Because the changes in mechanical and impact properties of irradiated specimens are highly dependent on the irradiation temperature, it is necessary to have knowledge of the temperature of specimens as well as the pressure vessel. During irradiation, instrumented capsules are not practical for a surveillance program extending over the design lifetime of a power reactor. The maximum temperature of the irradiated specimens can be estimated with reasonable accuracy by including within the capsule assembly small pieces of low melting point alloys or pure metals. The compositions of candidate materials with melting points in the operating range of power reactors are listed in Table 5.3-8. The monitors are selected to bracket the operating temperature range of the reactor vessel.

The temperature monitors consist of a helix of low melting alloy wire inside a sealed quartz tube. A stainless steel weight is provided to destroy the integrity of the wire when the melting point of the alloy is reached. The compositions and therefore the melting temperatures of the temperature monitors are differentiated by the physical lengths of the quartz tubes which contains the alloy wires.

A set of temperature monitors is included in each tensile-monitor compartment. The temperature monitors are placed in holes drilled in stainless steel housings as shown in Figure 5.3-7 and are also placed at three axial locations in each capsule assembly (Figure 5.3-5) to provide an axial profile of the maximum temperature to which the specimens were exposed.

5.3.1.6.3.3 Irradiation Locations

The encapsulated test specimens are irradiated at approximately identical radial positions about the midplane of the core. The test specimens are enclosed within six capsule assemblies the axial positions of which are bisected by the midplane of the core. A summary of the specimens contained in each of these capsule assemblies is presented in Table 5.3-9.

The test specimens contained in the capsule assemblies are used for monitoring the neutron-induced property changes of the reactor vessel materials. These capsules, therefore, are positioned near the inside wall of the reactor vessel so that the irradiation conditions (fluence, flux spectrum, temperature) of the test specimens resemble as closely as possible the irradiation conditions of the reactor vessel. The neutron fluence of the test specimens is expected to be approximately 50 percent greater than that seen by the adjacent vessel wall.

The RT_{NDT} changes resulting from the irradiation of these specimens closely approximate the RT_{NDT} changes in the materials of the reactor vessel.

The capsule assemblies are placed in capsule holders positioned circumferentially about the core at locations which include the regions of maximum flux. Figure 5.3-8 shows the location of the capsule assemblies.

All capsule assemblies are inserted into their respective capsule holders during the final reactor assembly operation.

5.3.1.6.3.4 Capsule Assembly Removal

The capsule assemblies remain within their capsule holders until the test specimens contained therein have attained desired levels of exposure (EFPY). At that time, selected capsule assemblies are removed. The distribution of target exposures for removal of capsule assemblies is presented in Table 5.3-10.

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The target exposure levels for the surveillance capsules are based on the time intervals indicated in the withdrawal schedule in ASTM E-185-82, referenced in 10CFR50, Appendix H.

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Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns which most closely approach the withdrawal schedule.

During unit start-up and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon a rate of 100° F/hr and for cyclic operation.

The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section III of the ASME Code including Appendix G, Protection Against Nonductile Failure and the rules contained in 10CFR50, Appendix G, Fracture Toughness Requirements. Compliance with the criteria in 10CFR50, Appendix H is discussed in Subsection 5.3.1.6.

5.3.1.7 Reactor Vessel Fasteners

The stud material for the reactor vessel closure head is fabricated from SA 540 Grade B24 Class III material. The nuts and washers for the reactor vessel fasteners are made from SA 540 Grade B24 or B23 material. These materials were ordered prior to the issuance of 10CFR50, Appendix G and Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs". Material tests for Waterford 3 stud material demonstrates adequate toughness in accordance with ASME Code Section III, 1971 Edition through Summer 1971 Addenda, and an acceptable level of ultimate tensile strength which is consistent with the recommendations of Reg Guide 1.65 (see Table 5.3-11). Test results demonstrate that the stud material meets the 25 mil lateral expansion, 45 ft-lb criteria of 10CFR50, Appendix G at 10°F and that they do not exceed 170 KSI ultimate tensile strength (UTS).

Testing adequate to establish compliance with the ASME Code Section III, 1971 Edition through Summer 1971 Addenda, was done for the nut and washer material. 10CFR50, Appendix G, Section IV, "Fracture Toughness Requirements", Paragraph 4 requires that material for bolting and other fasteners with nominal diameters exceeding one in. shall meet the minimum requirements of 25 mils lateral expansion and 45 ft-lbs in terms of Charpy V-notch tests conducted at the preload temperature or at the lowest service temperature, whichever is lower.

In order to determine whether the nuts and washers met the 10CFR50 Appendix G requirements, all Charpy test data for SA540 Gr B-24 steel (Southern California Edison Electric Co.'s San Onofre Generating Station Units 2 and 3 and Waterford Unit 3 reactor vessel studs, washers and nuts) was accumulated. Waterford Unit 3 data is presented in Table 5.3-12 and all the data is graphically presented in Figure 5.3-9 (Charpy Energy Absorbed) and Figures 5.3-10 (Mils Lateral Expansion). It can be seen that the available data plotted in Figures 5.3-9 indicates that a lower bound curve through minimum points yields a Charpy absorbed energy value of approximately 46 ft-lbs at +60°F. Further, a similar lower bound curve in Figure 5.3-10 yields a value of 27 mils lateral expansion at +60°F. Since both these curves are minimum point curves for data points from six separate heats, the temperature necessary to meet 10CFR50, Appendix G requirements (C_V = 45 ft-lbs, lateral expansion = 25 mils) is 60°F.

5.3.2 PRESSURE TEMPERATURE LIMITS

5.3.2.1 Limit Curves

The reactor vessel beltline material consists of six plates. The nil ductility transition temperatures (T_{NDTT}) of each plate was established by drop weight test. Charpy tests were then performed to determine at what temperature the plates exhibited 50 ft-lb absorbed energy and 35 mils lateral expansion. From this testing a reference temperature for transverse direction (RT_{NDT}) of 22°F was established.

For the remaining material in the RCS, a limiting RT_{NDT} of 90°F was established based upon SA 105 Class 2 material used to fabricate the lower driver mount flanges of the reactor coolant pumps.

→(DRN 00-1059, R11-A)

As a result of fast neutron irradiation in the region of the core, RT_{NDT} will increase with operation. The techniques used to analytically and experimentally predict the integrated fast neutron ($E \geq 1$ MeV) fluxes of the reactor vessel are described in Subsections 5.3.1.4 and 5.3.1.6. Extent of compliance with Regulatory Guide 1.99 is discussed in Subsection 5.3.1.4.

→(DRN 03-2059, R14)

Since the neutron spectra and flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured reference transition temperature shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron flux variation. The peak end-of-license (32 EFPY) neutron fluence ($E > 1.0$ MeV) at the core midplane for the Waterford Unit 3 reactor vessel is 2.48×10^{19} n/cm^2. That is the fluence corresponding to the clad-base metal interface. Projections of neutron fluence beyond Cycle 11 were based on a 1.5% uprate (3441 MWt) at the start of Cycle 12 and a 8% uprate (3716 MWt) at the start of Cycle 14. The highest predicted Adjusted Reference Temperature (ART) at 32 EFPY is 50°F[7]. This corresponds to an integrated fast neutron fluence ($E > 1.0$ MeV) at the ¼ thickness of 1.48×10^{19} n/cm^2 and was determined using the methodology of Regulatory Guide 1.99, Revision 2. The actual shift in RT_{NDT} will be established periodically during plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Subsection 5.3.1.6 and shown in Figure 5.3-8. To compensate for any increase in the RT_{NDT} caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. During the first 10 years of reactor operation, a conservatively high fluence of 9.2×10^{18} n/cm^2 is assumed which corresponds to 3580 Mwt and 80 percent load factor. The corresponding ΔRT_{NDT} is 75°F based on the curve shown in Figure 5.3-1. Thus, for this interval, the upper limit to the RT_{NDT} is (initial + shift) or 22°F + 75°F = 97°F. This is greater than (and conservatively bounds) the 40 years Adjusted Reference Temperature at 88°F calculated from the methodology of Regulatory Guide 1.99, Revision 2. The limit lines identified in Technical Specification 16.3/4.4 are based on the following:

←(DRN 00-1059, R11-A; 03-2059, R14)

a) Heatup and Cooldown Curves (from Section III of the ASME Code Appendix G-2215)

K_{IR} = $2 K_{IM} + K_{IT}$

K_{IR} = Allowable stress intensity factor at temperatures related to RT_{NDT} (ASME III Figure G-2110-1)

K_{IM} = Stress intensity factor for membrane stress (pressure) The 2 represents a safety factor of 2. on pressure

K_{IT} = Stress intensity factor for radial thermal gradient

The above equation is applied to the reactor vessel beltline.

For plant heatup the thermal stress varies from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e. no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

For plant cooldown thermal and pressure stress are additive. The design cooldown rate of 100°F/hr is used for calculation.

$K_{IM} = M_M \frac{PR}{t}$

M_M = ASME III, Figure G-2214-1

P = Pressure, psi.

R = Vessel radius - in.

t = Vessel wall thickness - in.

K_{IT} = $M_T \; T_W$

M_T = ASME III, Figure G-2214-2

T_W = Highest radial temperature gradient through wall at end of cooldown.

K_{IT} is therefore calculated at a maximum- gradient and is considered a constant = A for cooldown and zero for heatup. $M_M R/T$ is also a constant = B.

therefore:

$K_{IR} = BP + A$

$$P = \frac{K_{IR} - A}{B}$$

→(DRN 06-842, R15)

K_{IR} is then varied as a function of temperature from Figure G-2110-1 of ASME III and the allowable pressure calculated. Other regions of the reactor vessel have also been analyzed. With the exception of the vessel flange during heatup, the beltline region is controlling after considering RT_{NDT} shifts. The limit curve for heatup is a composite of the limitations imposed by the vessel flange and beltline region. Instrumentation errors and hydrostatic head corrections are considered when implementing the curves. Whenever the core is critical, an additional 40°F is added to these curves as required by 10CFR50, Appendix G.

←(DRN 06-842, R15)

b) System In-service Testing

The in-service testing curve is developed in the same manner as in a) above with the exception that a safety factor of 1.5 is allowed by ASME III in lieu of 2.

c) Lowest Service Temperature

As indicated previously, an RT_{NDT} for all material with the exception of the reactor vessel beltline was established at 90°F. ASME III, Art. NB-2332 (b) require a lowest service temperature of RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, a pressure of 20 percent of the system hydrostatic test pressure cannot be exceeded.

d) Maximum Pressure for Shutdown Cooling

This pressure is established by considering the design pressure of the Shutdown Cooling System, shutoff head of the low pressure safety injection (LPSI) pumps, elevation head from the pressurizer to the LPSI pumps and the design temperature of the Shutdown Cooling System.

The pressure-temperature limitation curves are predicted for 40-year life. During plant life operation, the surveillance capsules (refer to Subsection 5.3.3.7) will be removed from their location in the reactor vessel for testing. The data obtained will be compared to that used to develop the predicted limitation curves presented in the Technical Specifications.

If this information indicates anomalies to the existing predictions, the curves will be redrawn as previously indicated to reflect actual data.

5.3.2.2 Operating Procedures

→(DRN 06-911, R15)

Pressure-temperature limitations and additional information are described in the Technical Specifications. The pressure-temperature limit curves provided in Section 3/4.4 have been prepared in accordance with Appendix G, ASME Code Section III. Maintenance of reactor coolant system (RCS) pressure and temperature within these prescribed limits ensure that the integrity of the reactor coolant pressure boundary (RCPB) is maintained.

←(DRN 06-911, R15)

5.3.2.3 Fracture Toughness for Pressurized Thermal Shock Events

→(DRN 00-1059, R11-A; 03-2059, R14)

An evaluation was performed of the Waterford Unit 3 reactor vessel beltline materials relative to the Pressurized Thermal Shock (PTS) screening criteria of 10CFR50.61[3] and the upper shelf screening criteria of 10 CFR Part 50, Appendix G[4]. The PTS values are calculated in accordance with 10CFR50.61[3]. The predicted upper shelf energy values are evaluated using the methods of Regulatory Guide 1.99, Revision 2[5], for each beltline material. The calculation of the PTS and upper shelf energy values represents power uprate conditions, including a 1.5% uprate (3441 MWt) at the start of Cycle 12 and a 8% uprate (3716 MWt) at the start of Cycle 14.

The determination of the chemistry factor values per Position 1.1 and 2.1 of Reference 5 is detailed in Reference 6 and summarized in Table 5.3-14. (In this table, the “Chemistry Factor Basis” refers to values from Table 1 and 2 of 10CFR50.61[3] for weld and plate, respectively, and to “surveillance data” for the values derived in Reference 6.) The neutron fluence is the peak value for the corresponding plate and weld for 32 EFPY. RT_{PTS} was determined for each material in the beltline region is given by the following expression:

$$RT_{PTS} = \text{Initial } RT_{NDT} + \Delta RT_{PTS} + \text{Margin}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. Measured values of initial RT_{NDT} are available for each of the materials. ΔRT_{PTS} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{PTS} = CF * f^{(0.28 - 0.10 \log f)}$$

where f is the vessel fluence at the clad-base metal interface given in units of 10^{19} n/cm^2. Margin is determined based on the uncertainty in initial RT_{NDT} and the uncertainty in the ΔRT_{PTS} prediction. Margin is calculated as:

$$M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2}$$

The initial RT_{NDT} values are based on measured values and, therefore, σ_i is equal to 0°F. The uncertainty in the ΔRT_{PTS} prediction, σ_Δ, is 28°F for welds and 17°F for plates. However, the value of $2\sigma_\Delta$ does not have to exceed ΔRT_{PTS}. (Note: The value of σ_Δ for the two materials for which credible surveillance data are available does not have to exceed 14°F for welds and 8.5°F for base metal). The values of σ_i, σ_Δ, and the total margin are given for each material in Table 5.3-14. Total margin in these cases is $2\sigma_\Delta$ or ΔRT_{PTS}, whichever is smaller. Values of RT_{PTS} are given for each material in Table 5.3-14. The highest value is 53°F for lower shell plate M-1004-2 at 32 EFPY. All the projected values for the Waterford Unit 3 reactor vessel beltline materials are well below the Pressurized Thermal Shock (PTS) screening criteria of 270°F for axial welds and plates, and 300°F for circumferential welds.

←(DRN 00-1059, R11-A; 03-2059, R14)

➔(DRN 03-2059, R14)
The predicted upper shelf energy values for each of the Waterford 3 Unit beltline material were evaluated in accordance with 10 CFR Part 50, Appendix G[4] using Position 1.2 of Regulatory Guide 1.99, Revision 2[5]. The predictions were based on the predicted fluence[6] at the vessel ¼T location at 32 EFPY. (Note: The peak beltline fluence at ¼T was used for all the beltline materials rather than taking credit for the relatively small variation in the axial fluence between the intermediate and lower shells.) The projected upper shelf energy values far exceed the 50 ft-lb screening criterion of 10CFR50, Appendix G[4] at 32 EFPY.

The plate and weld data from the surveillance capsule analyses[6] were also evaluated using Position 2.2 of Regulatory Guide 1.99, Revision 2[5]. The projections of upper shelf energy decrease were based on the upper bound of the measurements extended parallel to the lines in Figure 2 of Reference 5. The projected shelf decrease for plate M-1004-2 based on the measurements is 14% at 32 EFPY. The projected shelf decrease for weld 101-171 based on the measurements is 9.5% at 32 EFPY. The projected upper shelf energies are given below at 32 EFPY. The corresponding values based on Position 1.2 are also shown below. This comparison demonstrates the low radiation sensitivity of the Waterford Unit 3 beltline materials, and it adds confidence to the expectation that those materials will far exceed the 50 ft-lb screening criterion of 10CFR50, Appendix G[4] at 32 EFPY.

<u>Comparison of Upper Shelf Energy Decrease Predictions</u>

Material	**32 EFPY USE (ft-lbs) using Position 1.2**	**32 EFPY USE (ft-lbs) using Position 2.2**
Plate M-1004-2	117	121
Weld 101-171	124	141

🡐(DRN 03-2059, R14)

5.3.3 REACTOR VESSEL INTEGRITY

➔(EC-1020, R307)
C-E designed and fabricated the reactor vessel for Waterford 3 C-E has been involved in reactor vessel design and fabrication since the late 1950's, and this proven expertise is reflected in the Waterford 3 reactor vessel and the satisfactory performance of a large number of reactor vessels in operating plants. Westinghouse designed the Replacement Reactor Vessel Closure Head (RRVCH). The RRVCH was fabricated by Doosan.
🡐(EC-1020, R307)

Vessel integrity is ensured by the use of proven fabrication techniques and well characterized steels which exhibit uniform properties and consistent behavior. The characterization of these materials was established through industrial and governmental studies which examined the prefabrication material properties through to irradiated service operation. Inservice inspection and material surveillance programs are also conducted during the service life of the vessel, which further ensures that vessel integrity is maintained.

5.3.3.1 <u>Design</u>

Applicable design codes are found in Table 5.2-1. A schematic of the reactor vessel is shown on Figure 5.3-11. Additional information can be found in Subsection 5.3.1.2.

5.3.3.2 Materials of Construction

The reactor vessel shell is fabricated from SA-533, Grade B, Class 1, material. This material has a minimum tensile strength of 80 ksi and a minimum yield strength of 50 ksi. This shell material responds well to quench and tempered heat treatment, which in combination with fine-grain melting practice produces high quality plate with excellent fracture toughness properties. The nozzles, also having excellent toughness properties, are fabricated from SA508, Class 2 forgings. The welding materials used include Mil Spec B-4 wire for submerged arc processes and E-8018C-3 material for manual arc processes. The stainless steel cladding utilized is nominal 19Cr-9Ni.

5.3.3.3 Fabrication Methods

➔(EC-1020, R307)
The reactor vessel is constructed of formed plates welded into cylinders and hemispherical heads. The closure head, upper shell, and nozzles are forgings. This typifies construction of the reactor vessel in the preceding introductory material. No special fabrication methods were used in the reactor vessel fabrication. The basic design and fabrication of the reactor vessel are as follows.

The reactor vessel is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical upper closure head. The pressure vessel is approximately 520 in. high (overheads) by 172 in. inside diameter and is all welded manganese molybdenum nickel steel plate and forging construction. Except for the Replacement Reactor Vessel Closure head which is clad with 3/16 in. minimum. Type 308L (and 309L used for base layer) stainless steel, the internal surfaces that are in contact with the reactor coolant are clad with 1/8 in. minimum Type 304 austenitic stainless steel and have a finish of 250 micro in. or better. The closure head flange and reactor vessel shell flange provide the structural rigidity necessary for bolting the head to the shell.
🡐(EC-1020, R307)

The reactor vessel fabrication is begun with an upper vessel assembly which consists of the upper shell, intermediate shell, nozzles, and reactor vessel shell flange. Both the upper and intermediate shells consist of three 120 degree segments formed from plate material and welded together to form cylindrical shells. Once the shells are welded, the upper shell is welded to the reactor vessel shell flange. The intermediate shell is then welded to form the upper vessel assembly. Four inlet nozzles and two outlet nozzles are then welded to complete the upper vessel assembly.

The lower vessel assembly consists of the lower shell and the bottom head. The lower shell is formed from three plates into 120 degree segments and welded together to form a cylindrical shell. The bottom head is constructed of six peel segments and a dome section, all formed from plate material. These are welded together to form a hemispherical head. The lower shell and bottom head are then welded together to complete the lower vessel assembly.

The closure head is fabricated separately. It is bolted to the reactor vessel only for hydrostatic testing.

➔(EC-1020, R307)
The closure head assembly consists of the closure head forging, control element drive mechanism (CEDM) housings, and instrument nozzles. Penetrations are then machined in the closure head for 87 control rod mechanisms, 10 instrumentation nozzles, and one vent pipe. Attachment of these complete the closure head assembly. The closure head is attached to the reactor vessel by 54 seven in. diameter studs which are threaded into the vessel flange and extend through the closure head flange.
🡐(EC-1020, R307)

Previous experience using the above procedures in fabricating other reactor vessels is summarized in Subsection 5.3.3.

5.3.3.4 Inspection Requirements

→(EC-1020, R307)

Inspection requirements of ASME Code, Section III, 1971 Edition including Summer 1971 Addenda and, for the replacement closure head, Section III, 1998 Edition through 2000 Addenda, are discussed in Subsection 5.3.1.3.

←(EC-1020, R307)

5.3.3.5 Shipment and Installation

→(EC-1020, R307)

The reactor vessel is shipped by barge to the site mounted on the shipping skid used for installation. The vessel is protected by closing all openings (including the top of the vessel) with metal shipping covers and pressurizing with inert gas. The replacement closure head is shipped on a separate skid. During shipment, the environment within the replacement closure head is maintained clean and dry, and is protected from external humidity and atmosphere by its shipping skid, shrink wrap and the use of desiccants. Vessel surfaces and covers are sprayed with a strippable coating for protection against corrosion during shipping and installation. Prior to the welding of inter-connecting piping and installation of insulation, the temporary protective coating is removed by peeling.

←(EC-1020, R307)

5.3.3.6 Operating Conditions

Operating parameters are provided in Subsection 4.4.3. Design transient information is supplied in Subsection 3.9.1.1.

5.3.3.7 In-service Surveillance

5.3.3.7.1 Irradiated Materials Surveillance

This program is described in Subsection 5.3.1.6.

5.3.3-7.2 In-service Inspection

This program is described in Subsection 5.2.4.

SECTION 5.3: REFERENCES

1. Letter from A. E. Scherer (C-E) to Secretary of the Commission (NRC), LD-75-655, September 26, 1975.

2. "C-E Procedure for Design, Fabrication, Installation and Inspection of Surveillance Specimen Holder Assemblies," Combustion Engineering Topical Report, CENPD-155P, September 1974.

→(DRN 03-2059, R14)

3. 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” Federal Register, Volume 60, No. 243, dated December 19, 1995.

4. Code of Federal Regulations, 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No.243, dated December 19, 1995.

5. Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” U.S. Nuclear Regulatory Commission, May 1988.

6. WCAP-16002, Revision 0, “Analysis of Capsule 263 from the Entergy Operations Waterford Unit 3 Reactor Vessel Radiation Surveillance Program,” March 2003.

7. WCAP-16088, Revision 1, “Waterford Unit 3 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation,” September 2003.

→(DRN 03-2059, R14)

TABLE 5.3-1 (Sheet 1 of 5)

DATA POINTS USED TO ESTABLISH C-E RT_{NDT} SHIFT vs. FLUENCE AND PERCENT C UPPER SHELF DROP vs. (FLUENCE) 1/2 DESIGN CURVES FOR A533-B REACTOR VESSEL MATERIALS

Data Point	Lit Ref[h]	Type Mtl	Plate or Weld	Thickness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	ϕt (n/cm^2 x 10^{19})[a]	$(\phi t)^{1/2}$ $(n/cm^2)^{1/2}$ x 10^9	ΔRT_{NDT}[b] (°F)	C_v Initial Upper Shelf (ft-lb)	C_v Post-Irradiation Upper Shelf (ft-lb)	C_v Upper Shelf Drop (ft-lb)	Percent C_v Upper Shelf Drop (%)
32	1	A533-B1	Plate	4	0.14	0.009	0.022	2.3	4.79	120	100	77	23	23
33	1	A533-B1	Plate	8	0.14	0.010	0.023	2.3	4.79	95	116	98	18	15.5
34	1	A533-B1	Plate	8-1/8	0.19	0.010	0.017	1.7	4.12	190	101	86	15	14.8
35	1	A533-B1	Plate	6-3/8	0.09	0.008	0.015	0.2	1.41	0	137	137	0	0
36	1	A533-B1	Plate	6-3/8	0.09	0.008	0.015	2.0	4.47	80	137	130	7	5.1
37	1	A533-B1	Plate	6-3/8	0.09	0.008	0.015	2.0	4.47	90	137	75	-	-
38	1	A533-B2	Plate	6-3/8	0.09	0.008	0.015	0.5	2.23	35	120	-	-	-
39	1	A533-B2	Plate	6-3/8	0.09	0.008	0.015	2.0	4.47	75	120	70	50	-
40	1	A533-B1	Plate	7-1/2	0.12	0.008	0.015	1.7	4.12	70	136	100	36	-
41	1	A533-B1	Plate	7-1/2	0.11	0.008	0.019	1.7	4.12	85	126	121	5	3.9
42	1	A533-B1	Plate	5-3/4	0.12	0.008	0.018	1.8	4.24	50	148	115	33	22.3
43	2	A533-B1	Plate	8	0.09	0.008	0.014	0.5	2.23	0	135	135	0	0
44	2	A533-B1	Plate	8	0.09	0.008	0.014	2.4	4.89	85	135	125	10	7.4
45	3	A533-B1	Plate	6-1/4	0.09	0.003	0.014	2.5	5.0	60	123	128	-	-
48	1	A533-B1	Weld-S/A[d]	7-1/2	0.22	0.015	0.011	1.7	4.1	200	109	63	46	42.2
49	1	A533-B1	Weld-E/S[e]	5-3/4	0.19	0.008	0.014	1.6	4.0	165	82	79	3	3.6

a. Fluence: Neutron energies > I Mev, irradiation temperature - 550 °F
b. Based on NDTT measured at the C_v 30 ft-lb level
C. The weld from which these specimens were taken (S/A) was back chipped and rewelded probably with a manual arc. The specimens were taken from different areas of the weld so the chemistries of the specimen could tend to vary greatly.
d. S/A = Submerged arc
e. E/S = Electroslag
f. Percent C_v , upper shelf drop as reported in reference
g. Irradiation temperature - 530 °F
h. Refer to Table 5.3-2

TABLE 5.3-1 (Sheet 2 of 5)

DATA POINTS USED TO ESTABLISH C-E RT_{NDT} SHIFT vs. FLUENCE AND PERCENT C_v UPPER SHELF DROP vs. (FLUENCE) 1/2 DESIGN-CURVES FOR A533-B REACTOR VESSEL MATERIALS

Data Point	Lit Ref[h]	Type Mtl	Plate or Weld	Thickness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	ϕt (n/cm^2 x 10^{19})[a]	$(\phi t)^{1/2}$ (n/cm^2)$^{1/2}$ x 10^9	ΔRT_{NDT}[b] (°F)	C_v Initial Upper Shelf (ft-lb)	C_v Post-Irradiation Upper Shelf (ft-lb)	C_v Upper Shelf Drop (ft-lb)	Percent C_v Upper Shelf Drop (%)
50[c]	2	A533-B1	Weld-S/A[d]	8	0.09	0.010	0.014	0.5	2.23	0	145	145	0	0
51[c]	2	A533-B1	Weld-S/A[d]	8	0.09	0.010	0.014	2.4	4.89	90	145	125	20	13.8
52[c]	2	A533-B1	Weld-S/A[d]	8	0.14	0.010	0.014	0.5	2.2	105	105	70	35	33.3
53[c]	2	A533-B1	Weld-S/A[d]	8	0.14	0.010	0.014	2.4	4.8	210	105	65	40	38
54	3	A533-B1	Weld-E/S[e]	6-1/4	0.09	0.002	0.012	2.5	5.0	100	88	78	10	11.3
58	4	A533-B1	Plate (surf.)	12	0.18	0.008	0.008	1.0	-	101	-	-	-	-
59	4	A533-B1	Plate (1/2 T)	12	0.18	0.008	0.008	1.0	-	126	-	-	-	-
60	4	A533-B1	Plate (3/8 T)	12	0.18	0.008	0.008	1.0	-	70	-	-	-	-
61	5	A533-B1	Plate	12	0.14	0.008	0.016	4.52-5.59	7.0	80	120	100	20	16.6
62	5	A533-B1	Plate	12	0.14	0.008	0.016	3.64-4.24	6.32	135	>120	~108	-	-
63	5	A533-B1	Plate	12	0.14	0.008	0.016	1.18-1.33	3.6	85	>120	~110	-	-
64	5	A533-B1	Weld-(S/A)[d]	11- 3/4	0.22	0.019	0.13	2.73-4.25	5.91	256	115	~60	55	47.8

TABLE 5.3-1 (Sheet 3 of 5)

DATA POINTS USED TO ESTABLISH C-E RT_{NDT} SHIFT vs. FLUENCE AND PERCENT C_v

UPPER SHELF DROP vs. (FLUENCE) 1/2 DESIGN CURVES FOR A533-B REACTOR VESSEL-MATERIALS

Data Point	Lit Ref(h)	Type Mtl	Plate or Weld	Thickness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	ϕt (n/cm^2 x 10^{19})(a)	$(\phi t)^{1/2}$ $(n/cm^2)^{1/2}$ x 10^9	ΔRT_{NDT}(b) (°F)	C_v Initial Upper Shelf (ft-lb)	C_v Post-Irradiation Upper Shelf (ft-lb)	C_v Upper Shelf Drop (ft-lb)	Percent C_v Upper Shelf Drop (%)
65	6	A533-B1	Plate	6	0.03	0.008	0.008	2.8	-	65	-	-	-	-
66	6	A533-B1	Plate	6	0.03	0.008	0.008	2.8	-	40	-	-	-	-
67	7	A533-B1	Plate	12	0.18	0.008	0.008	0.47	-	70	-	-	-	-
68	7	A533-B1	Plate	12	0.18	0.008	0.008	0.94	3.06	95	104	100	4	3.8
69	7	A533-B1	Plate	12	0.18	0.008	0.008	1.05	-	130	-	-	-	-
70	8	A533-B1	Weld-S/A(d)	12	0.23	0.011	0.008	2.5	5.0	270	125	70	55	44
71	8	A533-B1	Plate	12	0.18	0.008	0.008	2.8	5.29	200	104	73	31	29.8
72	9	A533-B1	Plate	6	0.13	0.008	0.008	2.8	5.29	125	135	100	35	25.9
73	9	A533-B1	Plate	6	0.13	0.008	0.007	2.8	5.29	140	110	90	20	13.18
74	9	A533-B1	Plate	6	0.03	0.008	0.008	3.1	5.56	70	145	138	7	4.8
75	10	A533-B1	Plate (3/8 T)	12	0.14	0.008	0.016	0.5	-	50	-	-	-	-
77	8	A533-B1	Plate	12	0.14	0.008	0.016	2.7	5.19	170	122	~102	20	14(f)
78	8	A533-B1	Plate	12	0.14	0.008	0.016	2.6	5.09	165	99	~85	14	14(f)
79	11	A533-B1	Plate	8-10	0.17	0.009	0.015	2.1	4.58	145	115	93	22	19.1
80	11	A533-B1	Plate	8-10	0.24	0.008	0.011	3.7	6.08	165	110	84	26	23.6
81	11	A533-B1	Weld-(S/A)(d)	8-10	0.36	0.015	0.012	3.4	5.83	315	107	56	51	47.6

TABLE 5.3-1 (Sheet 4 of 5)

DATA POINTS USED TO ESTABLISH C-E RT_{NDT} SHIFT vs. FLUENCE AND PERCENT C_v

UPPER-SHELF DROP vs. (FLUENCE) 1/2 DESIGN-CURVES FOR A533-B REACTOR VESSEL MATERIALS

Data Point	Lit Ref[h]	Type Mtl	Plate or Weld	Thick ness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	ϕt (n/cm^2 x 10^{19})[a]	$(\phi t)^{1/2}$ (n/cm^2)$^{1/2}$ x 10^9	ΔRT_{NDT}[b] (°F)	C_v Initial Upper Shelf (ft-lb)	C_v Post-Irradiation Upper Shelf (ft-lb)	C_v Upper Shelf Drop (ft-lb)	Percent C_v Upper Shelf Drop (%)
82	11	A533-B1	Weld S/A[d]	8-10	0.20	0.016	0.013	3.4	5.83	95	129	98	31	24.03
83	11	A533-B1	Plate	8-10	0.17	0.009	0.015	6.7	8.18	210	115	94	21	18.2
84	11	A533-B1	Plate	8-10	0.24	0.008	0.011	6.1	7.8	185	110	86	24	21.8
85	11	A533-B1	Weld-S/A[d]	8-10	0.36	0.015	0.012	6.1	7.8	350	107	56	51	47.6
86	11	A533-B1	Weld-S/A[d]	8-10	0.20	0.016	0.013	6.1	7.8	125	129	98	31	24.03
87	11	A533-B1	Plate	8-10	0.09	0.009	0.017	4.4	6.63	35	104	107	-	-
88	11	A533-B1	Plate	8-10	0.09	0.009	0.017	5.7	7.54	55	104	111	-	-
89	11	A533-B1	Plate	8-10	0.09	0.011	0.018	4.0	6.32	45	119	130	-	-
90	11	A533-B1	Plate	8-10	0.09	0.011	0.018	5.4	7.34	85	119	130	-	-
91	11	A533-B1	Weld-S/A[d]	8-10	0.07	0.010	0.010	4.9	7.0	35	157	155	2	1.27
92	11	A533-B1	Weld-S/A[d]	8-10	0.07	0.010	0.010	5.0	7.07	50	157	155	2	1.27
93	11	A533-B1	Weld-S/A[d]	8-10	0.05	0.004	0.004	4.9	7.0	≤20	144	147	-	-
94	12	A533-B1 (same material as Pt. 74)	Plate	6	0.03	0.008	0.008	15.8[g]	12.56	260	145	99	46	31.7

TABLE 5.3-1 (Sheet 5 of 5)
DATA POINTS USED TO ESTABLISH C-E RT_{NDT} SHIFT vs. FLUENCE AND PERCENT C_v
UPPER SHELF DROP vs. (FLUENCE) 1/2 DESIGN CURVES FOR A533-B REACTOR VESSEL MATERIALS

Data Point	Lit Ref[h]	Type Mtl	Plate or Weld	Thick ness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	ϕt (n/cm^2 x 10^{19})[a]	$(\phi t)^{1/2}$ (n/cm^2)$^{1/2}$ x 10^9	ΔRT_{NDT}[b] (°F)	C_v Initial Upper Shelf (ft-lb)	C_v Post-Irradiation Upper Shelf (ft-lb)	C_v Upper Shelf Drop (ft-lb)	Percent C_v Upper Shelf Drop (%)
95	12	A533-B1 (same material as Pt. 74)	Plate	6	0.03	0.008	0.008	24.1(g)	15.52	335	145	70	75	51.7
96	12	A533-B1 (same material as Pt. 74)	Plate	6	0.03	0.008	0.008	27.8	16.67	295	145	99	46	31.7
97	13	A533-B1 (same material as Pt. 73)	Plate	6	0.13	0.008	0.007	0.095	-	0	-	-	-	-
98	13	A533-B1 (same material as Pt. 85)	Weld	8-10	0.36	0.015	0.018	0.095	-	55	-	-	-	-
A	14	A302-B	Weld	10-1/2	0.22	-	-	0.2	-	95	-	-	-	-
B	15	A302-B	Weld	6	0.27	0.014	0.012	0.7	-	140	-	-	-	-
C	16	A508-2	Weld	6-1/2	0.23	0.012	0.016	0.49	-	140	-	-	-	-
D	1	A533-2	Weld	4	0.27	0.016	0.015	1.4	-	205	-	-	-	-

TABLE 5.3-2 (Sheet 1 of 2)

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TABLE 5.3-2 (Sheet 2 of 2)

LITERATURE REFERENCES FOR TABLE 5.3-1

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13. Naval Research Laboratory, unpublished data.

14. Ireland D. R., and Scotti, V. G., "Final Report on Examination and Evaluation of Capsule A for the Connecticut Yankee Reactor Pressure Vessel Surveillance Program," Battelle (Columbus) Memorial Institute, Docket No. 50-213, 1970.

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17. Smidt, F. A., Jr. and Sprague, J. A., "Property Changes Resulting from Impurity Defect Interaction in Iron and Pressure Vessel Steel Alloys," Effects of Radiation on Substructure and Mechanical Properties of Metals and Alloys, ASTM STP 529, pp 78-91, 1973.

TABLE 5.3-3

SUMMARY OF SURVEILLANCE MATERIALS TESTING

Material and Code	C_V Upper Shelf (ft-lb)	30 ft-lb Fit [a] (°F)	50 ft-lb Fit [b] (°F)	35 Mils Lat. Exp. Fit (b) (°F)	NDTT (°F)	RT_{NDT} (°F)	RT Yield Strength (ksi) Static	Dynamic
Base Metal Plate M-1004-2 (WR)	136	- 30	18	-2	-20	-20	69	97
Base Metal Plate M-1004-2 (RW)	169.5	0	48	36	0	--	70	103
Weld Metal M-1004-1/M-1004-3	146	- 76	-46	-46	-80	-80	85	113
HAZ Metal M-1004-2	163.5	-106	-72	-76	-50	-50	70	113
SRM HSST Plate O1MY-RW	130	28	70	48	0	--	--	--

(a) Determined from average impact energy curve.

(b) Determined from lower bound curve.

TABLE 5.3-4 Revision 15 (03/07)

TOTAL QUANTITY OF SPECIMENS

Type of Specimen	Orientation	Base Metal	Weld Metal	HAZ	SRM[a]	Totals
Drop Weight →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	12	____	____	____	12
	Transverse	12	12	12	----	36
Charpy Impact →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	78	---	---	39	117
	Transverse	102	102	102	---	306
Tensile →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	18	---	---	---	18
	Transverse	36	36	36	---	108
→(DRN 06-897, R15) Totals ←(DRN 06-897, R15)		258	150	150	39	597

(a) Standard Reference Material

TABLE 5.3-5 Revision 15 (03/07)

TYPE AND QUANTITY OF-SPECIMENS FOR UNIRRADIATED TESTS

Type of Specimen	Orientation	Quantity of Specimens Base Metal	Weld Metal	HAZ	SRM[a]	Totals
Drop Weight →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	12	---	---	---	12
	Transverse	12	12	12	---	36
Charpy Impact →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	30	---	---	15	45
	Transverse	30	30	30	---	90
Tensile →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	18	---	---	---	18
	Transverse	18	18	18	---	54
→(DRN 06-897, R15) Totals ←(DRN 06-897, R15)		120	60	60	15	255

(a) Standard Reference Material characterized by HSST Program, specimens provided only for correlation with characterization tests.

TABLE 5.3-6 Revision 15 (03/07)

TYPE AND QUANTITY OF SPECIMENS FOR

IRRADIATION EXPOSURE AND IRRADIATED TESTS

Type of Specimen	Orientation	Quantity of Specimens				
		Base Metal	Weld Metal	HAZ	SRM[a]	Totals
→(DRN 06-897, R15) Charpy Impact	Longitudinal	48	-- --	24	--	72
←(DRN 06-897, R15)	Transverse	72	72	72	--	216
Tensile	Longitudinal	--	-- --	--	--	
→(DRN 06-897, R15) ←(DRN 06-897, R15)	Transverse	18	18	18	--	54
→(DRN 06-897, R15) Totals ←(DRN 06-897, R15)		138	90	90	24	342

a. Standard Reference Material

TABLE 5.3-7

CANDIDATE MATERIALS FOR NEUTRON THRESHOLD DETECTORS

Material	Reaction	Threshold Energy (MeV)	Half-Life
Uranium	U^{238} (n, f) Sr^{90}	0.7	28 years
Sulfur	S^{32} (n, p) P^{32}	2.9	14.3 days
Iron	Fe^{54} (n, p) Mn^{54}	4.0	314 days
Nickel	Ni^{58} (n, p) Co^{58}	5.0	71 days
Copper	Cu^{63} (n,α) Co^{60}	7.0	5.3 years
Titanium	Ti^{46} (n, p) Sc^{46}	8.0	84 days
Cobalt	Co^{59} (n,γ) Co^{60}	Thermal	5.3 years

TABLE 5.3-8

COMPOSITION AND MELTING POINTS OF

CANDIDATE MATERIALS FOR TEMPERATURE MONITORS

Composition (wt%)	Melting Temperature (°F)
80 Au, 20 Sn	536
90.0 Pb, 5.0 Sn, 5.0 Ag	558
97.5 Pb, 2.5 Ag	580
97.5 Pb, 0.75 Sn, 1.75 Ag	590

TABLE 5.3-9 Revision 13-A (09/04)

TYPE AND QUANTITY OF SPECIMENS CONTAINED IN EACH

IRRADIATION CAPSULE ASSEMBLY

→(DRN 04-1049, R13-A)

Capsule Location	Base Metal Impact L[b]	Base Metal Impact T[c]	Base Metal Tensile	Weld Metal Impact	Weld Metal Tensile	HAZ Impact	HAZ Tensile	Reference Impact[a]	Total Specimens Impact	Total Specimens Tensile
Vessel-97°	12	12	3	12	3	12	3	--	48	9
Vessel-104°	--	12	3	12	3	12	3	12	48	9
Vessel-284°	12	12	3	12	3	12	3	--	48	9
Vessel-263°	--	12	3	12	3	12	3	12	48	9
Vessel-277°	12	12	3	12	3	12	3	--	48	9
Vessel-83°	12	12	3	12	3	12	3	--	48	9
Totals	48	72	18	72	18	72	18	24	288	54

←(DRN 04-1049, R13-A)

a. Reference material correlation monitors
b. L = Longitudinal
c. T = Tranverse

TABLE 5.3-10 Revision 14 (12/05)

CAPSULE ASSEMBLY REMOVAL SCHEDULE

→(DRN 03-2059, R14)

Capsule No./ID	Azimuthal Location (deg)	Lead Factor	Removal Time (EFPY)*	Target Fluence (n/cm^2)
1/W-83	83	1.18	26	2.47 x 10^{19}
2/W-97	97	1.18	4.44**	6.47 x 10^{18}**
3/W-104	104	0.83	Standby	--
4/W-263	263	1.18	13.83**	1.45** x 10^{19}**
5/W-277	277	1.18	Standby	--
6/W-284	284	0.83	Standby	--

←(DRN 03-2059, R14)

*EFPY - Effective Full Power Years, withdrawal time may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.
** - Values represent actual data on removed capsule

→(DRN 04-1049, R13-A)

NOTE: As required by 10CFR Appendix H, Section III.B.3, submit a proposed withdrawal schedule with technical justification as specified in 10CFR50.4 for NRC approval prior to implementation.

←(DRN 04-1049, R13-A)

TABLE 5.3-11

WATERFORD UNIT 3 REACTOR VESSEL CLOSURE STUDS DATA

Piece Number	Drawing Number	Code Number	Heat No.	Material	Ultimate Tensile Strength Test Temp (°F)	Strength (KSI)	Test Temp (°F)	Fracture Toughness Charpy Energy (ft-lbs.)	Mils Lateral Expansion
98	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	168	10	49-48-47	28-27-25
98-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	162.5	10	50-51-51	25-28-29
69	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	163	10	50-48-49	26-25-29
69-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	157	10	57-56-57	38-35-37
70	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	156	10	56-54-53	30-34-34
70-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	164	10	50-50-48	34-29-31
72	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	158	10	56-56-55	38-34-32
72-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	159	10	50-50-51	26-25-28
74	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	157	10	50-52-50	31-32-27
74-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	158	10	50-51-50	25-25-30
76	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	154	10	54-54-53	27-31-33
76-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	161	10	50-50-51	25-30-29

TABLE 5.3-12

WATERFORD UNIT 3 REACTOR VESSEL NUTS AND WASHERS DATA

					Ultimate Tensile Strength		Fracture Toughness		
Piece Number	Drawing Number	Code Number	Heat No.	Material	Test Temp (F)	Strength (KSI)	Test Temp. (F)	Charpy Energy (Ft-lbs)	Mils Lateral Expansion -
41	E74170-161-03	M-1029-1	18551	SA-540 Grade B23	70	163.5	10	38-40-38	19-21-18
41-1	E74170-161-03	M-1029-1	18551	SA-540 Grade B23	70	164.5	10	42-40-38	20-22-18
48	E74170-161-03	M-1029-1	18551	SA-540 Grade B23	70	170.0	10	37-39-38	18-19-21
48-1	E74170-161-03	M-1029-1	18551	SA-540 Grade B23	70	165.0	10	43-45-42	25-27-24

WATERFORD UNIT 3
REACTOR VESSEL MATERIALS

Product Form	Material Identification	Drop Weight NDTT (°F)	Initial[d] RT_{NDT} (°F)	Chemical Content %		
				Nickel	Copper	Phosphorus
Plate	M-1003-1	-30	-30	0.71	0.02	0.004
Plate	M-1003-2	-50	-50	0.67	0.02	0.006
Plate	M-1003-3	-50	-42	0.70	0.02	0.007
Plate	M-1004-1	-50	-15	0.62	0.03	0.006
Plate	M-1004-2	-20	22	0.58	0.03	0.005
Plate	M-1004-3	-50	-10	0.62	0.03	0.007
Weld	101-124 A,B,& C[a]	-60	-60	0.96	0.02	0.010
Weld	101-142 A,B,& C[b]	-80	-80	< 0.20	0.03	0.007
Weld	101-171 c	-70 to-80	-70 to-80	0.16	0.05	0.008

a. Intermediate shell course longitudinal seam weld

b. Lower shell course longitudinal seam weld

c. Intermediate - lower shell girth weld

d. Plate RT_{NDT} determined using Branch Technical Position MTEB 5-2; weld RT_{NDT} determined in accordance with ASME Code, Section III, NB-2300

TABLE 5.3-14 Revision 14 (12/05)

CALCULATION OF THE WATERFORD UNIT 3 RT_{PTS} VALUES FOR 32 EFPY

Material	Chemistry Factor Basis	CF (°F)	Fluence ($x10^{19} n/cm^2$)	RT_{NDT} [(a)] (°F)	ΔRT_{PTS} [(b)] (°F)	σ_i (°F)	σ_Δ (°F)	Margin (°F)	RT_{PTS} [(c)]
Intermediate Shell Plate M-1003-1	Table 2	20	2.48	-30	24.9	0	12.4	24.9	20
Intermediate Shell Plate M-1003-2	Table 2	20	2.48	-50	24.9	0	12.4	24.9	0
Intermediate Shell Plate M-1003-3	Table 2	20	2.48	-42	24.9	0	12.4	24.9	8
Lower Shell Plate M-1004-1	Table 2	20	2.47	-15	24.9	0	12.4	24.9	35
Lower Shell Plate M-1004-2	Surveillance Data	12.4	2.47	22	15.4	0	7.7	15.4	53
Lower Shell Plate M-1004-3	Table 2	20	2.47	-10	24.9	0	12.4	24.9	40
Intermediate Shell Longitudinal Weld Seams 101-124 A,B,C	Table 1	27	2.48	-60	33.6	0	16.8	33.6	7
Lower Shell Longitudinal Weld Seams 101-142 A,B,C	Table 1	35	2.47	-80	43.5	0	21.8	43.5	7
Intermediate to Lower Shell Girth Weld Seam 101-171	Surveillance Data	16.2	2.47	-70	20.1	0	10.1	20.1	-30

<u>Notes:</u>

(a) Initial reference temperature (RT_{NDT}) values are measured. Thus, σ_i equal to 0°F.
(b) ΔRT_{PTS} = CF * FF
(c) RT_{PTS} = Initial RT_{NDT} + ΔRT_{PTS} + Margin (°F)

←(DRN 03-2059, R14)