4.2 REACTOR VESSEL AND APPURTENANCES MECHANICAL DESIGN

4.2.1 Power Generation Objective

The reactor vessel power generation design objective is to provide a volume in which the core can be submerged in coolant, thereby allowing power operation of the fuel. The reactor vessel and appurtenances design provides the means for the attachment of pipelines to the reactor vessel and the means for the proper installation of vessel internal components.

4.2.2 Power Generation Design Basis

1. The location and design of the external and internal supports provided as an integral part of the reactor vessel shall be such that stresses in the reactor vessel and supports due to reactions at these supports are within ASME Boiler and Pressure Vessel Code limits.

2. The reactor vessel design lifetime shall be 40 years. Time Limited Aging Analyses (TLAAs) have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Section O.3.1 and O.3.2.

3. The design of the reactor vessel and appurtenances shall allow for the accomplishment of a suitable program of periodic inspection and surveillance.

4.2.3 Safety Design Basis

1. The reactor vessel and appurtenances shall be designed to withstand adverse combinations of loadings and forces resulting from operation under abnormal and accident conditions.

2. The reactor vessel shall be designed and fabricated to a high standard of quality to provide assurance of an extremely low probability of failure.

3. To minimize the possibility of brittle fracture failure of the nuclear system process barrier, the following shall be required: (1) the initial ductile-brittle transition temperature of materials used in the reactor vessel shall be known by reference or established empirically; (2) expected shifts in transition temperature during design service life due to environmental conditions, such as neutron flux, shall be determined and employed in the reactor vessel design; and (3) operation margins to be observed with regard to the transition temperature shall be designated for each mode of operation.

4. The design shall provide for material surveillance specimens which may be used to verify predicted radiation exposure and to measure the effect of radiation on the vessel material.

4.2.4 Description

4.2.4.1 Reactor Vessel

The reactor vessel is a vertical, cylindrical pressure vessel with hemispherical heads of welded construction. The reactor vessel is designed and fabricated for a useful life of 40 years based upon the specified design and operating conditions. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2. The vessel for each unit is designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 edition, Summer 1965 addenda (Unit 3 vessel - Summer 1966 addenda), code cases 1332-1, 1332-2, 1332-3, 1334, 1335-2 (paragraph 4), 1336, 1339, applicable requirements for Class A vessels as defined therein, and additional GE requirements. The reactor vessel and its supports are designed as Class I equipment in accordance with the loading criteria of Appendix C. The materials used in the design and fabrication of the reactor pressure vessel are shown in Table 4.2-1.

The Browns Ferry Unit 1 vessel was fabricated by B&W. The Browns Ferry Units 2 and 3 vessels were fabricated by Ishikawajima-Harima Heavy Industries Co. (IHI) in Japan, under a contract between B&W and IHI. IHI had previously manufactured the Fukushima I and II vessels. These vessels are built to the ASME Boiler and Pressure Vessel Code and GE specifications. Reactor vessel data is presented in Table 4.2-2.

The cylindrical shell and bottom hemispherical head of the reactor vessel are fabricated of low alloy steel plate which is clad on the interior with weld overlay. The cylindrical shell is clad with stainless steel, and the bottom hemispherical head is clad with Inconel. The plates and forgings are ultrasonically tested and magnetic-particle-tested over 100 percent of their surfaces after forming and heat treatment. Full-penetration welds are used at all joints, including nozzles, throughout the vessel, except for nozzles of less than 3-inch nominal size and the CRD housing-to-stub tube welds. Nozzles of less than 3-inch nominal size are partial-penetration-welded as permitted by ASME Boiler and Pressure Vessel Code, Section III. The electroslag weld process was used on the Browns Ferry pressure vessels. Electroslag welding process variables, quality control procedures and technical details were presented in Appendix F, Dresden 2/3 FSAR, Docket Nos. 50-237 and 50-249.

Although little corrosion of plain carbon or low-alloy steels occurs at temperatures of 500°F to 600°F, higher corrosion rates occur at temperatures around 140°F. The 0.125-inch minimum-thickness cladding provides the necessary corrosion resistance during reactor shutdown and also helps maintain water clarity during refueling operations. Since the vessel head is exposed to a saturated steam environment throughout its operating lifetime, stainless steel cladding is not required over its interior surfaces. Exterior, exposed ferritic surfaces of pressure-containing parts have a minimum corrosion allowance of 1/16 inch. The interior surfaces of the top head and all carbon and low-alloy steel nozzles exposed to the reactor coolant have a corrosion allowance of 1/16 inch. The vessel shape is designed to limit coolant retention pockets and crevices.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle, rather than ductile, manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1×10^{17} nvt with neutrons of energies in excess of 1 MeV. Since the material NDT temperature dictates the minimum operating temperature at which the reactor vessel can be pressurized, it is desirable to keep the NDT temperature as low as possible. One way that this is accomplished is by selecting fine-grained steels and by using advanced fabrication techniques to minimize radiation effects. The as-fabricated initial NDT temperature for all carbon and low-alloy steel used in the main closure flanges, closure bolting material, and the shell and head materials connecting to these flanges, including the connecting circumferential weld material, is limited to a maximum of 10°F as determined by ASTM E208. For each main closure flange forging, a minimum of 1 tensile, 3 Charpy V-notch, and 2 drop weight test specimens have been tested from each of two locations about 180° apart on the flange. For all other carbon and low-alloy steel pressure-containing materials, including weld materials and the vessel support skirt material, the initial NDT temperature is no higher than 56°F for Unit 1, and 40°F for Units 2 and 3. A grain size of 5 or finer, as determined by the method in ASTM E112, is maintained.

Another way of minimizing any changes (elevating) to the NDT temperature is by reducing the integrated neutron exposure at the inner surface of the reactor vessel. The coolant annulus between the vessel and core shroud and the core location in the vessel limit the integrated neutron exposure of reactor vessel material to less than 1×10^{19} nvt from neutrons with energy levels greater than 1 MeV, within the 40-year design lifetime of the vessel. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2. This is not the expected exposure, nor is it the absolute limit of safe exposure; it is an exposure value that can be demonstrated to be safe and practical to maintain. The maximum calculated exposure for neutrons of 1 MeV or greater is 1.58×10^{18} nvt for Unit 1, per GEH Report No. 0000-0166-0632-R0. The maximum calculated exposure for neutrons of 1 MeV or greater is 1.93 X 10^{19} nvt for Unit 2, per GEH Report No. 000N2175-R1. The maximum calculated exposure for neutrons of 1 MeV or greater is 2.23 X 10^{18} nvt for Unit 3, per GEH Report No. 000N2183-R0. These maximum calculated exposures encompass the Browns Ferry unit power history since initial operation as well as a conservative prediction of future Browns Ferry operation up to 3952 MWt within the MELLL domain.

Operation in the MELLLA+ domain results in enhanced spectral shift during the operating cycle which results in more top-peaked axial power shape/flux shape. In addition, the MELLLA+ operating domain expansion results in a slightly higher operating neutron flux in the upper portion of the reactor core due to decreased water density. The net effect of spectral shift and water density reduction is a small increase in peak flux above the active fuel. Refinements in the calculations to support MELLLA+, documented in GEH Report No. NEDC-33877P, shows that the previous calculated exposures for Units 1, 2, and 3 from neutrons of 1 MeV or greater remain bounding.

The vessel top head is secured to the reactor vessel by studs and nuts which are designed to be tightened with a stud tensioner. The vessel flanges are sealed by two concentric metallic seal-rings designed for no detectable leakage through the inner or outer seal at any operating condition, including: (a) cold hydrostatic pressure test at the hydro-pressure specified in the ASME code, and (b) heating to operating pressure and temperature at a maximum rate of 100°F/hr. To detect lack of seal integrity, a 1-inch vent tap is provided in the area between the two seal-rings, and a monitor line is attached to the tap to provide an indication of leakage from the inner seal-ring seal (see Subsection 7.8). A 1-inch tap is also provided in the area outside the outer seal-ring for use in monitoring leakage. This tap is used only if the inner seal fails and is piped to an accessible place in the drywell and capped.

The head and vessel flanges are low-alloy steel forgings. The sealing surfaces of the reactor vessel head and shell flanges are weld-overlay clad with Inconel 82 (ERNiCr material. The clad thickness is 0.25 inches on both the head flange and shell flange sealing surfaces.

All sensitized austenitic stainless steel has been replaced on the Browns Ferry pressure vessels, except the jet pump riser brace pads on all units. These components have been clad with nonfurnace-sensitized stainless steel weld overlay. Austenitic stainless steel used in other component parts of the reactor coolant pressure boundary, including relief and safety valves, is fully annealed to preclude sensitization.

Welding processes were limited to 110,000 joules per inch and the interpass temperature limited to 350°F to avoid local sensitization of stainless steel.

Stainless steel with deliberate additions of nitrogen for enhancing the material strength has not been used.

The vessel nozzles (Figure 4.2-2) are low-alloy steel forgings made in accordance with ASTM A508 CL2 as modified by ASME code case 1332-2, paragraph 5. Nozzles of 3-inch nominal size or larger are full-penetration welded to the vessel. Nozzles of less than 3-inch nominal size may be partial-penetration-welded as permitted by ASME Boiler and Pressure Vessel Code, Section III. Nozzles which are partial-penetration welded are nickel-chromium-iron forgings made in accordance with ASME SB166 as modified by code case 1336.

The vessel top head nozzles are provided with flanges with small groove facing. The drain nozzle is of the full-penetration weld design and extends 16 inches below the bottom outside surface of the vessel. The recirculation inlet nozzles are located as shown in Figures 4.2-1, 4.2-3, and 4.2-4; feedwater inlet nozzles, core spray inlet nozzles, and the control rod drive hydraulic system return nozzle have thermal sleeves similar to those shown in the detail of Figure 4.2-2.

As a result of cracks discovered in the feedwater nozzle blend and nozzle bore regions of several operating reactors, General Electric and the NRC performed an extensive study of the problem. The program, the solutions, and NRC acceptance of the modifications are fully described in NEDE 21821-A, "Boiling Water Reactor Feedwater Nozzle/Sparger - Final Report," February 1980 (proprietary version). The modifications to the BFNP feedwater nozzles included: (1) removal of the stainless-steel-clad and heat affected zone of the feedwater nozzle bore and nozzle bend radius, and (2) machining the safe end and nozzle bore and inner bend radius to accept the improved double piston ring seal, interference fit spargers with forged tee design, and orificed elbow discharges. Implementing these modifications increased the assurance of maintaining vessel integrity by minimizing the potential for crack initiation due to thermal cycling.

The nozzle for the core differential pressure and standby liquid control pipe is designed with a transition so that the stainless steel outer pipe of the differential pressure and standby liquid control line (see Subsection 3.3, "Reactor Vessel Internals Mechanical Design") can be socket-welded to the inner end of the nozzle and so that the inner pipe passes through the nozzle. This design provides an annular region between the nozzle and the inner standby liquid control line to minimize thermal shock effects on the reactor vessel in the event that use of the Standby Liquid Control System is required.

The jet pump instrumentation penetration seal is welded directly to the outer end of the jet pump instrumentation nozzle. The stainless steel recirculation loop piping (see Subsection 4.3, "Reactor Recirculation System") is welded to the outer end of the recirculation outlet nozzle. The main steam line piping is welded to the outer end of the steam outlet nozzle. The piping attached to the vessel nozzle is designed, installed, and tested in accordance with the requirements of USAS B31.1.0, 1967 edition and the applicable GE design and procurement specifications, which were implemented in lieu of the outdated B31 Nuclear Code Cases-N2, N7, N9, and N10.

Thermocouple pads are located on the exterior of the vessel (see Table 4.2-3). At each thermocouple location, two 3/4-inch-diameter pads are provided: an end pad to hold the end of a 3/16-inch-diameter thermocouple and a clamp pad equipped with a set screw to secure the thermocouple.

The reactor vessel is laterally and vertically supported and braced to make it as rigid as possible without impairing the movements required for thermal expansion. Where thermal requirements prohibit the use of rigid supports, spring anchors or hydraulic snubbers are employed to resist earthquake forces, while allowing sufficient flexibility for thermal expansion.

4.2.4.2 Shroud Support

The reactor vessel shroud support is a radial, cylindrical shell that surrounds the reactor core assembly and is designed so that stresses due to reactions at the shroud support are within ASME code, Section III, requirements for normal, upset, emergency, and faulted loading conditions. The design of the shroud support also takes into account the restraining effect of the components attached to the support, their weight, and earthquake loadings. The vessel shroud support and other internal attachments (jet pump riser support pads, diffuser brackets, guide rod brackets, steam dryer support brackets, dryer holddown brackets, feedwater sparger brackets, and core spray brackets) are as shown in Figures 4.2-1, 4.2-3, and 4.2-4.

4.2.4.3 Reactor Vessel Support Assembly

The reactor vessel support assembly consists of a ring girder, sole plates, and the various bolts, shims, and set screws necessary to position and secure the assembly between the reactor vessel support skirt and the support pedestal. The concrete and steel support pedestal is constructed integrally with the building foundation. Steel anchor bolts are set in the concrete with the threads extending above the surface. The sole plates are set flat and level on the concrete, and the lower flange of the ring girder is set on top of the sole plates and shimmed as necessary to level the ring girder. The anchor bolts extend through both the sole plates and the ring girder bottom flange. High strength bolts are used to bolt the flange of the reactor vessel support skirt to the top flange of the ring girder. The ring girder and sole plates are fabricated of ASTM A36 structural steel according to AISC specifications.

4.2.4.4 Vessel Stabilizers

The vessel stabilizers are connected between the reactor vessel and the top of the shield wall surrounding the vessel to provide lateral stability for the upper part of the vessel. Eight stabilizer brackets are attached by full-penetration welds to the reactor vessel at evenly spaced locations around the vessel below the flange. Each vessel stabilizer consists of a stabilizer rod, threaded at the ends, springs, washers, nut, a plate, and a bumper bracket with tapered shims. The stabilizers are attached to each bracket and apply tension in opposite directions. The stabilizers are evenly preloaded with tensioners to the values of the residual loads. The stabilizers are designed to permit radial and axial vessel expansion, to limit horizontal vibration, and to resist seismic and jet reaction forces.

4.2.4.5 Refueling Bellows

The refueling bellows form a seal between the reactor vessel and the surrounding primary containment drywell to permit flooding of the space (reactor well) above the vessel during refueling operations. The refueling bellows assembly (see Figures 4.2-1, 4.2-3, and 4.2-4) consists of a Type 304 stainless steel bellows, a backing plate, a spring seal, and a removable guard ring. The backing plate surrounds the outer circumference of the bellows to protect it and is equipped with a tap for testing and for monitoring leakage. The self-energizing spring seal is located in the area between the bellows and the backing plate and is designed to limit water loss in the event of a bellows rupture by yielding to make a tight fit to the backing plate when subjected to full hydrostatic pressure. The guard ring attaches to the assembly and protects the inner circumference of the bellows. The guard ring can be removed from above to inspect the bellows. The assembly is welded to the reactor bellows support skirt and the reactor well seal bulkhead plate. The reactor bellows support skirt is welded to the reactor vessel shell flange, and the reactor well seal bulkhead plate bridges the distance to the primary containment drywell wall. Six watertight, hinged covers are bolted in place for normal refueling operation. For normal operation, these covers are opened and removable air supply ducts and air return ducts permit circulation of ventilation air in the region above the reactor well seal.

4.2.4.6 Control Rod Drive Housings

The control rod drive housings are inserted through the control rod drive penetrations in the reactor vessel bottom head and are welded to the stub tubes extending into the reactor vessel[1] (Figure 4.2-2).

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1 Kobsa, I. R., and Wetzerl, V. R., "Design and Analysis of Control Rod Drive Reactor Vessel Penetrations," General Electric Co., Atomic Power Equipment Department, November 1968 (APED-5703).

Each housing transmits a number of loads to the bottom head of the reactor. These loads include the weight of a control rod and control rod drive, which are bolted to the housing from below (see Subsection 3.4, "Reactivity Control Mechanical Design"), the weight of a control rod guide tube, one four-lobed fuel support piece, and the four fuel assemblies which rest on the top of the fuel support piece (see Subsection 3.3, "Reactor Vessel Internal Mechanical Design"). The housings are fabricated of Type 304 austenitic stainless steel.

4.2.4.7 Control Rod Drive Housing Supports

The control rod drive housing support is designed to prevent a nuclear transient in the unlikely event that there is a control rod drive housing failure. This device consists of a grid structure located below the reactor vessel from which housing supports are suspended. The supports allow only slight movement of the control rod drive or housing in the event of failure. The control rod drive housing support is described in detail in Subsection 3.5, "Control Rod Drive Housing Supports."

4.2.4.8 In-Core Neutron Flux Monitor Housing

The in-core neutron flux monitor housings are inserted up through the in-core penetrations in the bottom head of the reactor vessel and are welded to the inner surface of the bottom head (Figure 4.2-2). An in-core flux monitor guide tube is welded to the top of each housing (see Subsection 3.3, "Reactor Vessel Internals Mechanical Design"), and either a source range monitor/intermediate range monitor (SRM/IRM) drive unit or a local power range monitor (LPRM) is bolted to the seal-ring flange at the bottom of the housing (see Subsection 7.5, "Neutron Monitoring System").

4.2.4.9 Reactor Vessel Insulation

The reactor vessel insulation is an all-metal, reflective insulation having an average maximum heat transfer rate of approximately 80 Btu/hr-ft^2 at the operating conditions of 550°F for the vessel and 135°F for the outside air. The maximum insulation thickness ranges from 4 inches for the upper head to 3-1/2 inches for the cylindrical shell and nozzles and 3 inches for the bottom head. The insulation is designed to permit complete submersion in water without loss of insulating material, contamination from the water, or adverse effect on the insulation efficiency of the insulation assembly after draining and drying. The lower head and cylindrical shell insulation is permanently installed for the 60 year operating life of the vessel. The insulation panels for the cylindrical shell of the vessel are held in place by vessel insulation supports located at two elevations on the vessel. The support brackets for each support are full-penetration-welded to the vessel at 12 evenly spaced locations around the circumference. Provisions are made for removing insulation during inservice inspection.

4.2.4.10 Other Reactor Coolant Pressure Boundary Ferritic Components

The fracture or notch-toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness when the system is pressurized to more than 20 percent of the design pressure. Such assurance is provided by maintaining the lowest service metal temperature, when the system pressure exceeds 20 percent of design pressure, at least 60°F above the nil-ductility transition temperature (NDTT). The lowest service-metal temperature is the lowest temperature which the metal will experience in service while the plant is in operation. It is established by appropriate calculations considering atmosphere ambient temperatures, the insulation or enclosure provided, and the minimum temperature maintained. Further interpretations and requirements are as follows:

A. Charpy V-notch (American Society for Testing and Material Standard A370 Type A) or drop weight (per ASTM E208) tests have been performed to demonstrate that all materials and weld metal meet brittle fracture requirements at test temperature. Test specimens, for the surveillance capsule pulled in 1994, were prepared and tested with minimum impact energy requirements in accordance with Table N-421 and the general provisions of N-313, N-331, N-332, and N-511 of Section III of the ASME Boiler and Pressure Vessel Code. For the surveillance capsule pulled in 2011, per BWRVIP-271/NP, the Charpy impact tests were conducted in accordance with ASTM Standards E185-82 and E23-02. Prior to the Summer 1972 Addenda of the 1971 ASME Section III Boiler and Pressure Vessel Code, impact testing was not required on materials with a nominal section thickness of 1/2 inch or less. However, this 1/2 inch thickness exclusion was increased to 5/8 inch by the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, Summer 1972 Addenda. Therefore, after issuance of the Summer 1972 Addenda, impact testing is not required on materials with a nominal section thickness of 5/8 inch or less. The welding procedures used were qualified by impact testing of weld metal and heat affected zone to the same requirements as the base metal in accordance with N-541.

B. Impact tests were not required for the following:

   1. Bolting, including nuts, 1-inch nominal diameter or less,

   2. Bars with a nominal cross-sectional area not exceeding 1 square inch,

   3. Materials with a nominal (section) wall thickness of less than 1/2 inch or 5/8 inch (refer to paragraph 4.2.4.10.A),

4. Components including pumps, valves, piping, and fittings with a nominal inlet pipe size of 6-inch-diameter and less, regardless of thickness, and

5. Consumable insert material, austenitic stainless steel, and nonferrous materials.

C. Impact testing was not required on components or equipment pressure parts having a minimum service temperature of 250°F or more when pressured over 20 percent of the design pressure. Example: Steam line is excluded from brittle fracture test requirement since the steam temperature will be over 250°F when the steam line pressure is at the 20 percent design pressure.

D. Impact testing was not required on components or equipment pressure parts whose rupture could not result in a loss of coolant exceeding the capability of normal makeup systems to maintain adequate core cooling for the duration of a reactor shutdown and orderly cooldown.

E. These criteria apply to components and equipment pressure parts, including flange bolts of the reactor coolant pressure boundary, and do not apply to related components such as anchors, anchor bolts, hangers, suppressors, and restraints.

All components for the Browns Ferry plant were designed and fabricated giving consideration to brittle-fracture control requirements as stated above. However, these specific conditions were not a part of the initial Browns Ferry Units 1 and 2 plant requirements, and due to the status of fabrication on two items, the requirements could not be imposed without scrapping all materials. On Browns Ferry Units 1 and 2 these two items are: (1) feedwater piping through the second containment isolation valve, and (2) the 14-inch HPCI testable check valve (HPCI pump return into feedwater pipe outside the containment). Charpy V-notch impact tests were performed on these items where possible, and results indicate they generally would not meet the conditions under A, above, if they had been imposed.

4.2.5 Safety Evaluation

The reactor vessel design pressure of 1250 psig is determined by an analysis of margins required to provide a reasonable range for maneuvering during operation, with additional allowances to accommodate transients above the operating pressure without causing operation of the safety valves. The design temperature for the reactor vessel (575°F) is based on the saturation temperature of water corresponding to the design pressure.

To withstand external and internal loadings while maintaining a high degree of corrosion resistance, a high-strength, carbon-alloy steel is used as the base metal with an internal cladding applied by weld overlay to the cylindrical shell and bottom head. Use of the ASME Boiler and Pressure Vessel Code, Section III, Class A, pressure vessel code design criteria provides assurance that a vessel designed, built, and operated within its design limits has an extremely low probability of failure due to any known failure mechanism.

The reactor vessel is designed for a 40-year life and will not be exposed to more than 1 X 10^{19} nvt of neutrons with energies exceeding 1 MeV. The reactor vessel is also designed for the transients which could occur during the 40-year life as indicated below. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2.

Type of Cycle	No. of Cycles
Boltup	123
Design hydrostatic test at 1250 psig	130
Startup (100°F/hr heatup rate)	120
Daily reduction to 75 percent power	10,000
Weekly reduction to 50 percent power	2,000
Control rod worth test	400
Loss of feedwater heaters (80 cycles total)	
Turbine trip at 25 percent power	10
Feedwater heater bypass	70
Scram (200 cycles total)	
Loss of feedwater pumps, isolation valves close	10
Turbine trip, feedwater on, isolation valves stay open	40
Reactor overpressure with delayed scram, feedwater stays on, isolation valves stay open	1
Single safety relief valve blowdown	2
All other scrams	147
Improper start of cold recirculation loop	5
Sudden start of pump in cold recirculation loop	5
Shutdown (100°F/hr cooldown rate)	118
Hydrostatic test at 1563 psig	3
Unbolt	123

Stress analysis and load combinations for the reactor vessel are evaluated for the cycles listed above, with the conclusion that ASME code limits are satisfied. The details of assumed loading combinations are described in Appendix C for Class 1 equipment. It is possible that the specified number of cycles for some of the events listed above may be exceeded over the life of the plant. A plant procedure has been implemented at Browns Ferry to maintain surveillance on the number of cycles which have occurred and the resulting fatigue usage factors. When the fatigue usage factor reaches a value of 0.7, the procedure requires a reevaluation to be completed in a timely manner to assure that the allowable fatigue usage factor of 1.0 is not exceeded. Operating limits on pressure and temperature during inservice hydrostatic testing were established using as a guide Appendix G to the ASME Boiler and Pressure Vessel Code, Section III, 1971, which was first added to the code in the summer 1972 addenda. The intent of Appendix G is to set criteria based on fracture toughness to provide a margin of safety against a nonductile failure. The resulting operating limits ensure that a large postulated surface flaw, having a depth of one-quarter of the material thickness and a length of one and one-half of the material thickness, can be safely accommodated in regions of the reactor vessel shell remote from discontinuities. Operating limits on temperature and pressure when the core is critical were established by using 10 CFR 50, Appendix G, "Fracture Toughness Requirements," paragraph IV.A.2.C. The 1998 Edition of the ASME Section XI Boiler and Pressure Vessel Code including 2000 Addenda was used in the development of the Unit 1 P-T curves. The P-T curve methodology includes the following: 1) the use of K_{1C} from Figure 4200-1 of Appendix A to Section XI and 2) the use of the M_m calculation in the ASME Code paragraph G.2214 of Appendix G to Section XI for a postulated defect normal to the direction of maximum stress. An exemption from specific requirements of 10 CFR Part 50, Appendix G is taken by use of ASME Code Case N-640 for Unit 2 and Unit 3. ASME Code Case N-640 permits the use of an alternative reference fracture curve K_{1c} for RPV materials for use in determining the PT limits. The PT limit curves based on the K_{1c} fracture toughness curve enhance overall plant safety by minimizing challenges to operators since requirements for maintaining a high vessel temperature during pressure testing are lessened. ASME Code Case N-588 methodology was also used as a basis for the PT curves. This code case permits the use of an alternative procedure for calculating applied stress intensity factors during normal operation and pressure test conditions due to pressure and thermal gradients for axial flaws. This methodology is incorporated into the ASME, Section XI Code, 1995 Edition, 1996 Addenda, which is the current code of record for the Unit 2 inservice inspection program. Since Unit 3 uses an earlier code of record for the inservice inspection program, Unit 3 implements the requirements of only the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix G to allow the use of the ASME Code Case N-588 methodology for PT curves. The operating limits are provided in the technical specifications for Browns Ferry. For the purpose of setting these operating limits, the initial RT_{NDT} (nil-ductility reference temperature) was determined from the impact test data taken in accordance with the requirements of the code to which the reactor vessels were designed and manufactured. The maximum NDT temperature allowed by the vessel specifications was 40°F. Although test data on beltline base material show lower NDT temperatures, an assumed RT_{NDT} of 40°F was used in the vessel beltline area, as well as the areas remote from the beltline because the generally accepted NDT temperature for electroslag welds used in the beltline longitudinal seams is 40°F.

The current operating limits on the pressure/temperature (P/T) curves in the technical specifications are based on the following (RT_{NDT}) values. Unit 1 has used 23.1°F for the (RT_{NDT}) value, Unit 2 has used 23.1°F for the (RT_{NDT}) value, and Unit 3 has used 23.1°F for the (RT_{NDT}) value.

For the current P/T curves, fluences were conservatively calculated for licensed operating periods of 38 EFPY for Unit 1, 48 EFPY for Unit 2, and 54 for Unit 3. These periods reflect 60-year reactor pressure vessel operating life and a conservative period of plant operation at 3952 MWt power level. The higher fluence was used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G in accordance with Regulatory Guide 1.99, Revision 2. The end-of-life shelf energy was evaluated by an equivalent margin analysis (EMA). The results of these evaluations indicated that:

(a) The results of the upper shelf energy EMA for limiting welds and plates for the three vessels remain less than the acceptance criterion in all cases.

(b) The effective full power year (EFPY) shift is slightly increased and, consequently requires a change in the adjusted reference temperature (ART), which is the initial RT_{NDT} plus the shift. The beltline material ART will remain within the 200 degree screening criterion.

In addition to the minimum requirements of the ASME Boiler and Pressure Vessel Code, the following precautions were taken and tests made either to ensure that the initial NDT temperature of the reactor vessel material is low or to reduce the sensitivity of the material to irradiation effects.

a. The material was selected and fabrication procedures were controlled to produce as fine a grain size as practical. It is an objective in fabrication to maintain a grain size of 5 or finer.

b. Drop weight impact tests were performed on each heat and heat treatment charge of all low-alloy steel-plate material in its "as-fabricated" condition.

c. Drop weight impact tests were made on the weld metal, the heat-affected zone of the base metal, and the base metal of the weld test plates simulating seams. If different welding procedures were used for nozzle welds, drop weight tests of similarly prepared coupons were made. The NDT temperature test criteria for the weld and heat-affected zone of the base material are the same as for the unaffected base metal.

d. The actual NDT temperature of the plates opposite the center of the reactor core was determined. In other areas it was sufficient to demonstrate that the two drop weight test specimens did not break at 10°F above the design NDT temperatures. The area of the vessel located opposite the core was fabricated entirely of plate and was not penetrated by nozzles, nor were there any other structural discontinuities in this area which would act as stress risers.

The reactor assembly is designed such that the average annular distance from the outermost fuel assemblies to the inner surface of the reactor vessel is approximately 80 centimeters. This annular volume, which contains the core shroud, the jet pump assemblies, and reactor coolant, serves to attenuate the fast neutron flux incident upon the reactor vessel wall. Using assumptions of plant operation at 3440 Mw(t), 100 percent plant availability, and 40-year plant life, the neutron fluence at the inner urface of the vessel was calculated to be 3.8 X 10^{17} nvt for neutrons having energies greater than 1 MeV. The results of the analyses of the vessel wall neutron dosimeters which were removed from the Browns Ferry reactor vessels at the end of the first core cycle indicated that the neutron fluence at the inner surfaces of the vessels at the end of 40-year plant life would be 1.56 x 10^{18}, 1.34 x 10^{18}, and 1.31 x 10^{18} nvt for Units 1, 2, and 3, respectively. These results ranged from 3-1/2 to 4 times the calculated fluence of 3.8 x 10^{17} nvt. Thus, additional analyses were required to predict the shifts in RT_{NDT} based on fluence obtained from the analyses of the vessel wall neutron dosimeters. The procedures in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 1, April 1977 were used to predict the RT_{NDT} shifts. Response to Generic Letter 92-01 provides updated fluence data. TLAAs have been identified and evaluated for the reactor vessel 60 year operating life. Summaries of these evaluations for the reactor vessel life are provided in Appendix O, Sections O.3.1 and O.3.2.

Quality control methods were used during the fabrication and assembly of the reactor vessel and appurtenances to ensure that the design specifications were met.

The fabrication test program was carried out by the reactor vessel vendors on material representative of the formed, heat-treated, and fully fabricated vessel. Tests of base metal and welded joint were performed, and the results were reported during the early stages of vessel construction. Tensile specimens (of 0.505 inch in diameter) from the shell plate material were prepared for various thickness levels of the plate material. These specimens were tested at various temperatures per ASTM Specifications E8 and E21 to determine tensile strength, yield strength, elongation, and reduction of area. Tensile specimens whose gauge diameter is at least 80 percent of the reactor vessel wall thickness were prepared from base metal and weld material. These specimens were tested at room temperature per ASTM Specification E8 to provide stress-strain curves, tensile strength, yield strength, elongation, reduction of area, and macrophotographs of the breaks. Charpy V-notch impact specimens were prepared from base metal and tested per ASTM Specification E23, Type A, to establish curves for determining the transition temperature at which 30 ft-lb of absorbed energy result in ductile fracture for various thickness levels of the plate material.

The Reactor Coolant System was cleaned and flushed before fuel was loaded initially. During the preoperational test program, the reactor vessel and Reactor Coolant System were given a hydrostatic test in accordance with code requirements at 125 percent of design pressure. The vessel temperature is maintained at a minimum of 60°F above the NDT temperature prior to pressurizing the vessel for a hydrostatic test. A hydrostatic test at a pressure not to exceed system operating pressure is made following each removal and replacement of the reactor vessel head. Other preoperational tests included calibrating and testing the reactor vessel flange seal-ring leakage detection instrumentation, adjusting reactor vessel stabilizers, checking all vessel thermocouples, and checking the operation of the vessel flange stud tensioner.

During the startup test program, the reactor vessel temperatures were monitored during vessel heatup and cooldown to assure that thermal stress on the reactor vessel was not excessive during startup and/or shutdown.

The average rate of reactor coolant temperature change during normal heatup and cooldown is limited to 100°F in any 1-hour period. Only during some postulated events, or in local areas, would this rate of fluid temperature change be exceeded as a result of rapid blowdown, valve operation, or rupture accident.

4.2.6 Inspection and Testing

The inservice inspection and testing program for the reactor vessel and appurtenances is outlined and detailed in Subsection 4.12. Extent and areas of examination, inspection methods, and frequency of examination are established therein.

The surveillance test program provides for the preparation of a series of Charpy V-notch impact specimens and tensile specimens from the base metal of the reactor vessel, weld heat-affected zone metal, and weld metal from a reactor steel joint which simulates a welded joint in the reactor vessel.

The reactor vessel material surveillance program is described in report NEDO-10115, Mechanical Property Surveillance of General Electric BWR Vessels, by J. P. Higgins and F. A. Brant. It describes the specimens, specimen inventory, capsule design, associated equipment, material selection and instructions for handling the specimens. All the requirements of paragraphs 3.1 through 3.3 of ASTM E-185-66 are met. All the requirements of paragraphs 4, 5, 6, 7, and 8 of ASTM E-185-66 are met, except that thermal control specimens discussed in paragraph 4.3 are not used. NEDO-10115, paragraph 5.7 states, "Because the BWR is a constant-temperature device, no special temperature monitoring devices are required." It is felt paragraph 4.3 of E-185-66 is a recommendation rather than a requirement.

The vessel surveillance samples were prepared in accordance with GE purchase specification 21A1111, Rev. No. 9, Attachment B.

The NDT temperatures for the three core region plates were as follows.

Heat No.	Plate No.	NDTT (F)
C2884-2	6-139-19	0
C2868-2	6-139-20	0
C2753-1	6-139-21	-20

The two test plates furnished by Babcock & Wilcox under the requirements of paragraph 3.1.1 of attachment B to specification 21A1111 were fabricated from Heat No. C2884-2 and C2868-2. The two plates were electroslag-welded (B&W Weld Procedure WR-12-4) and heat-treated the same as the core region plates. Tensile and Charpy impact specimen samples were removed as indicated in Figures 3, 4, 5, 6, and 7 of attachment B to 21A1111. (See FSAR Appendices J, K, and L.)

The surveillance test plate 610-0127 was 139 in. long and 60 in. wide, and all excess material is under TVA control in the event that additional material is needed. It is estimated that enough extra material is available for several hundred additional Charpy specimens.

No weak direction specimens were included in the reactor vessel material surveillance program. All Charpy V-notch specimens were taken parallel to the direction of rolling. The majority of developmental work on radiation effects has been with longitudinal specimens. This is considered the best specimen to be used for determination of changes in transition temperature. At the low neutron fluence levels of BWR plants, no change in transverse shelf level is expected and transition temperature changes are minimal.

The specimens and neutron monitor wires were placed near core midheight adjacent to the reactor vessel wall where the neutron exposure is similar to that of the vessel wall (see Subsection 3.3). The specimens were installed at startup or just prior to full-power operation. For Units 1, 2, and 3, Integrated Surveillance Program (ISP) implementation and surveillance specimen schedule withdrawal and testing is governed and controlled by BWRVIP-86 Revision 1-A, the BWRVIP responses to NRC RAIs dated May 30, 2001, December 22, 2001, and January 11, 2005, and the NRC's Safety Evaluation dated February 1, 2002. (NOTE: WRVIP-86, Revision 1-A, was approved by NRC and issues in October 2012, superseding both BWRVIP-86-A and BWRVIP-116.) Surveillance and chemistry data for all representative materials in the BWRVIP ISP have been consolidated into BWRVIP-135 {Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations.} A test specimen surveillance capsule (the second set of Unit 2 test specimens located at Azimuth 120º) was withdrawn in accordance with the ISP in 2011 during Unit 2 Refueling Outage 16 (U2R16) at approximately 23 EFPY of operation. An additional test specimen surveillance capsule is scheduled for withdrawal during the license renewal period, this being the third set of Unit 2 test specimens located at Azimuth 300º, which are currently scheduled for removal in the refueling outage closest to without exceeding 40 EFPY of operation. At the present time, this would correspond to Unit 2 Refueling Outage 24 (U2R24) in 2027. Presently, there are no plans to withdraw any capsules from either Unit 1 or Unit 3, as per BWRVIP-135, the BFN Unit 2 capsules provide the best representative plate material for all three units and the best representative weld material for Units 2 and 3. Supplemental Surveillance Program (SSP) Capsules A, B, D, G, E, and I provide the best representative weld material for Unit 1. Test results will provide the necessary data to monitor embrittlement for Units 1, 2, and 3. Since the predicted transition temperature shift of the reactor vessel beltline steel is less than 100°F at end-of-life, the use of the capsules per the ISP meets the requirements of 10 CFR 50, Appendix H, and ASTM E185-82. Revisions to fluence calculations using data obtained from the surveillance capsule specimens will use an NRC approved methodology that meets Regulatory Guide 1.190. {By letter dated August 14, 2008 (EDMS Number L44 080828 014), NRC issued License Amendment 273 for BFN Unit 1, and by letter dated January 28, 2003 (EDMS Number L44 030204 001), NRC issued License Amendment Numbers 279 and 238, for BFN Units 2 and 3 respectively, authorizing adoption of the BWRVIP Integrated Surveillance Program to address the requirements of Appendix H to 10 CFR Part 50.]