5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

5.3.1.1 Materials Specifications

The materials used in the reactor pressure vessel (RPV) and appurtenances are shown in Table 5.2-4, together with the applicable specifications.

The RPV materials shall comply with the provisions of ASME Code Section III, Appendix I, and meet the specification requirements of ASME Code Section II materials and 10CFR50 Appendix G.

5.3.1.2 Special Procedures Used for Manufacturing and Fabrication

The RPV is primarily constructed from low alloy, high-strength steel plate and forgings. Plates are ordered to ASME SA-533, TYPE B, Class 1, and forgings to ASME SA-508, Class 3. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low-alloy steels. Specific limits apply to materials used in the core beltline region of 0.05% maximum copper, 0.006% maximum phosphorous and 0.015% maximum sulfur in the base material and 0.05% maximum copper, 0.008% maximum phosphorus, 0.05% maximum vanadium and 0.015% maximum sulfur in the weld metal.

Studs, nuts, and washers for the main closure flange are ordered to ASME SA-540, Grade B23 or Grade B24. Welding electrodes for low alloy steel are low-hydrogen type ordered to ASME SFA-5.5, and weld filler metal to SFA-5.17.

All plate, forgings, and bolting are 100% ultrasonic (UT) examined and the surface examined by magnetic particle examination (MT) or liquid penetrant examination (PT) methods in accordance with ASME Code Section III, Division 1.

Fracture toughness properties are also measured and controlled in accordance with Division 1.

All fabrication of the RPV is performed in accordance with GE-approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates or forgings, and the flanges and nozzles from forgings. Welding performed to join these vessel components is in accordance with procedures qualified per ASME Code Section III and IX requirements. Weld test samples are required for each procedure for major vessel full-penetration welds. Tensile and impact tests are performed to determine the properties of the base metal, heat-affected zone, and weld metal.

Submerged arc and manual stick electrode welding processes are employed. Electroslag welding is not used for cladding. Preheat and interpass temperatures employed for welding of low-alloy steel meet or exceed the values given in ASME Code, Section III, Appendix D. Post-weld heat treatment at 593°C minimum is applied to all low-alloy steel welds.

Radiographic examination is performed on all pressure-retaining welds in accordance with requirements of ASME Code Section III, Subsection NB-5320. In addition, all pressure retaining welds are given a supplemental UT.

The materials, fabrication procedures, and testing methods used in the construction of the RPV meet or exceed requirements of ASME Code Section III, Class 1 vessels.

The cylindrical shells, top and bottom heads and main closure flanges of the reactor vessel are fabricated of low-alloy steel, the interior of which is clad with stainless steel weld overlay except for the top head, all nozzles but the steam outlet nozzles and the reactor internal pump penetration nozzles. The bottom head is clad with Ni-Cr-Fe alloy. The reactor internal pump casings are clad with Ni-Cr-Fe alloy, or alternatively stainless steel.

5.3.1.3 Special Methods for Nondestructive Examination

The materials and welds on the RPV are examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME Code Section III. The pressure-retaining welds are radiographically examined. In addition, the pressure-retaining welds are UT examined using acceptance standards that are equivalent or more restrictive than required by ASME Code Sections III and XI (see Paragraph 5.2.4.3.2.1). The UT examination method, including calibration, instrumentation, scanning sensitivity, and coverage, is based on the requirements imposed by ASME Code Section XI, Appendix I.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

5.3.1.4.1 Regulatory Guide 1.31: Control of Stainless Steel Welding

Controls on stainless steel welding are discussed in Subsection 5.2.3.4.2.1.

5.3.1.4.2 Regulatory Guide 1.34: Control of Electroslag Weld Properties

See Subsection 5.2.3.3.2.2.

5.3.1.4.3 Regulatory Guide 1.43: Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

Regulatory Guide 1.43 is concerned with the cracking of low-alloy steels underneath stainless steel weld deposited cladding. The requirements of this Regulatory Guide are not applicable to BWR vessels because: 1) the majority of the cladding is applied to low-alloy steel that is immune to underclad cracking; and 2) potentially susceptible low alloy steel is usually left unclad and is always in the fine grained condition, which renders it immune to underclad cracking.

5.3.1.4.4 Regulatory Guide 1.44: Control of the Use of Sensitized Stainless Steel

Sensitization of stainless steel is controlled by the use of service proven low carbon materials and by use of appropriate design and processing steps, including solution heat treatment, corrosion-resistant cladding, control of welding heat input, control of heat treatment during fabrication, and control of stresses.

5.3.1.4.5 Regulatory Guide 1.50: Control of Preheat Temperature for Welding Low-Alloy Steel

Regulatory Guide 1.50 delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

Low-alloy steel is used only in the reactor pressure vessel. Other ferritic components in the reactor coolant pressure boundary are fabricated from carbon steel materials.

Preheat temperature employed for welding of low alloy steel meet or exceed the recommendations of ASME Code Section III, Appendix D. Components are either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat is maintained until post-weld heat treatment. The minimum preheat and maximum interpass temperatures are specified and monitored.

Acceptance Criterion II.3.b(1)(a) of SRP Section 5.2.3 for control of preheat temperature requires that minimum and maximum interpass temperature be specified. While the Lungmen ABWR control of low-hydrogen electrodes to prevent hydrogen cracking (provided in Subsection 5.2.3.3.4) does not explicitly meet this requirement, the Lungmen ABWR control will assure that cracking of components made from low-alloy steels does not occur during fabrication. Further, the Lungmen ABWR control minimizes the possibility of subsequent cracking resulting from hydrogen being retained in the weldment.

All welds are nondestructively examined by radiographic methods. In addition, a supplemental UT examination is performed.

5.3.1.4.6 Regulatory Guide 1.71: Welder Qualification for Areas of Limited Accessibility

Qualification for areas of limited accessibility is discussed in Subsection 5.2.3.3.2.3.

5.3.1.4.7 Regulatory Guide 1.99: Effects of Residual Elements on Predicted Radiation Damage to Reactor Pressure Vessel Materials

Predictions for changes in transition temperature and upper shelf energy (USE) are made in accordance with the requirements of Regulatory Guide 1.99.

5.3.1.4.8 Regulatory Guide 1.37: Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

The cleaning of systems and components on the site during and at the completion of construction is accomplished to written procedures which assure both cleanliness and that the components are not exposed to materials or practices which will degrade their performance. For components containing stainless steel, the procedures will comply with Regulatory Guide 1.37. The procedures will prohibit contact with low melting point compounds, substances which are known to cause stress corrosion cracking or which can release in any manner substances that can cause such problems. In addition, there are controls placed on the use of grinding wheels and wire brushes that assure that they cannot introduce degrading materials either through prior usage or through their materials of construction (in this context, degradation includes stress corrosion cracking). Procedures are also used to prevent introduction of unnecessary dirt and require control of dirt producing processes such as welding or grinding including prompt cleaning.

5.3.1.5 Fracture Toughness

5.3.1.5.1 Compliance with 10CFR50, Appendix G

10CFR50 Appendix G is interpreted for Class 1 primary coolant pressure boundary component of the Lungmen ABWR reactor design and complied with as discussed in Subsections 5.3.1.5.2 and 5.3.2. The specific temperature limits on operation of the reactor when the core is critical are based on 10CFR50 Appendix G, Paragraph IV, A.3.

Fracture toughness data based on the limiting reactor vessel materials will be provided.

5.3.1.5.2 Methods of Compliance

The following items are the interpretations and methods used to comply with 10CFR50 Appendix G:

(1) Material Test Coupons and Test Specimens (GIII-A)

Test coupons are removed from the location in each product form as specified in Paragraph NB-2220 of ASME Code Section III. The heat treatment of the test coupons is performed in accordance with Paragraph NB-2210.

It is understood that separately produced test coupons per Subparagraph

NB-2223.3 may be used for forgings.

(2) Location and Orientation of Test Specimens (G III-A)

The test specimens are located and oriented per ASME Section III, Paragraph NB-2322. Transverse Charpy V-notch impact specimens are used for the testing of plate and forged material other than bolting and bars. Longitudinal specimens are used for bolting and bars.

Transverse specimens are used to determine the required minimum USE level of the core beltline materials.

In regard to 10CFR50 Appendix H, the surveillance test material is selected on the basis of the requirements of ASTM E185-82 and Regulatory Guide 1.99 to provide a conservative adjusted reference temperature for the beltline materials. The weld test plate for the surveillance program specimens has the principal working direction parallel to the weld seam to assure that heat-affected zone (HAZ) specimens are transverse to the principal working direction.

(3) Records and Procedures for Impact Testing (G III-C)

Preparation of impact testing procedures, calibration of test equipment, and the retention of the records of these functions and test data comply with the requirements of ASME Code Section III. Personnel conducting impact testing are qualified by experience, training or qualification testing that demonstrates competence to perform tests in accordance with the testing procedure.

(4) Charpy V-notch Curves for the RPV Beltline (G-IIIA and G-IVA-1)

A full transverse Charpy V-notch curve is determined for all heats of base material and weld metal used in the core beltline region with a minimum of three (3) specimens tested at the actual TNDT. The minimum USE level for base material and weld metal in the beltline region is 102.0 N•m as required by G-IVA.1.

In regard to G-III A, it is understood that separate, unirradiated baseline specimens per ASTM E-185, Paragraph 6.3.1 will be used to determine the transition temperature curve of the core beltline base material, HAZ and weld metal.

(5) Bolting Material

All bolting material exceeding 25.4 mm diameter has a minimum of 60.8 N•m Charpy-V energy and 0.64 mm lateral expansion at the minimum bolt preload temperature of 13°C.

(6) Alternative Procedures for the Calculation of Stress Intensity Factor (Appendix G-IV A)

Stress intensity factors are calculated by the methods of ASME Code Section III, Appendix G. Discontinuity regions are evaluated using the same general procedure as for shell and head areas. The evaluation is a part of the detailed thermal and stress analysis in the vessel stress report. Considerations are given to membrane and bending stresses, as outlined in Paragraph G-2222. Equivalent margins of safety to those required for shells and heads are demonstrated using a 0.25 T postulated defect at all locations, with the exception of the main closure flange to the head and shell discontinuity locations. Additional instruction on operating limits is required for outside surface flaw sizes greater than 6.0 mm at the outside surface of the flange to shell joint based on analysis made for Lungmen ABWR reactor vessels using the calculations methods shown in WRCB 175. It will be demonstrated, using a test mockup of these areas, that smaller defects can be detected by the ultrasonic inservice examinations procedures required at the adjacent weld joint.

(7) Fracture Toughness Margins in the Control of Reactivity (Appendix G-IV A).

10 CFR Appendix G and ASME Code Section III, Appendix G, will be used in determining pressure/temperature limitations for all phases of plant operation.

5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with Reactor Vessel Material Surveillance Program Requirements

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and thermal environment.

Reactor vessel materials surveillance specimens are provided in accordance with requirements of ASTM E-185 and 10CRF 50 Appendix H. Materials for the program are selected to represent materials used in the reactor beltline region. Charpy V-notch and tensile specimens are manufactured from the material actually used in the reactor beltline region. To represent those, if any, RPV pressure boundary welds that are in the beltline region (or are exposed to the predicted maximum neutron fluence ($E>1.6x10^{-13}J$) at the end of the design lifetime exceeding $1x10^{17}$ neutron/cm^2 at the inside surface of the reactor vessel), Charpy V-notch specimens of weld metal and HAZ material, and tensile specimens of weld metal are manufactured from the sample welds. The same heat of weld wire and lot of flux (if applicable) and the same welding practice as used for the beltline weld are utilized to make the sample welds. The specimen capsules are provided, each containing 12 Charpy V-notch and 3 tensile specimens of the beltline material and temperature monitors. Additionally, if required, the specimens identified to represent the welds requiring surveillance are also loaded in the same capsule. The surveillance specimen holders, brackets welded to the vessel cladding in the core beltline region, are provided to hold the specimen capsules and a neutron dosimeter. Since reactor vessel specifications require that all low-alloy steel pressure vessel boundary materials be produced to fine-grain practice, the bracket welding does not pose a concern of underclad cracking. A set of out-of-reactor baseline Charpy V-notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. The neutron dosimeter and temperature monitors will be located as required by ASTM E-185.

Six surveillance capsules are provided. The following will be identified for each surveillance capsule: (1) the specific materials in each surveillance capsule; (2) the capsule lead factors; (3) the withdrawal schedule for each surveillance capsule; (4) the neutron fluence to be received by each capsule at the time of its withdrawal; and (5) the vessel end-of-life peak neutron fluence. The predicted end of life of the adjusted reference nil ductility temperature of the reactor vessel steel is less than 5°C*.

The following proposed withdrawal schedule is extrapolated from ASTM E-185.

- First Capsule: After 6 effective full-power years.
- Second Capsule: After 15 effective full-power years.
- Third Capsule: With an exposure not to exceed the peak end of life (EOL) fluence.
- Fourth Capsule: Schedule determined based on results of first two capsules per ASTM E-185, Paragraph 7.6.2 . Testing of irradiated capsule specimens will be in accordance with requirements of ASTM E-185 as called out for by 10CFR50 Appendix H.
- Two additional capsules are provided per TPC request. These are not required by ASTM E-185 or by 10CFR50 Appendix H.

In addition, a block of material shall be installed in the capsules in order to provide irradiated archive material. The dimensions of this block will be as large as feasible but no less than 115 mm height minimum, with the same cross section as the capsule.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsections 4.1.4.5 and 4.3.2.2.

5.3.1.6.3 Predicted Irradiation Effects on Beltline Materials

Transition temperature changes and changes in upper-shelf energy shall be calculated in accordance with the rules of Regulatory Guide 1.99. Reference temperatures shall be established in accordance with 10CFR50 Appendix G and NB-2330 of the ASME Code.

Since weld material chemistry and fracture toughness data are not available at this time, the limits in the purchase specification were used to estimate worst-case irradiation effects.

These estimates show that the adjusted reference temperature at end-of-life is less than 5°C*, and the end-of-life USE exceeds 6.7 N•m (see response to Question 251.5 for the calculation and analysis associated with this estimate).

* Estimated values. The final values to be provided in the FSAR.

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment Appendix H.II B (2)

The surveillance specimen holders, described in Subsections 5.3.1.6.1 and 3.9.5.1.2.10, are located at different azimuths at common elevation in the core beltline region. The locations are selected to produce lead factor of approximately 1.2 to 1.5 for the inserted specimen capsules. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel. The capsules can be removed from and reinserted into the surveillance specimen holders.

In areas where brackets (such as the surveillance specimen holder brackets) are located, additional nondestructive examinations are performed on the vessel base metal and stainless steel weld-deposited cladding or weld-buildup pads during vessel manufacture. The base metal is ultrasonically examined by straight-beam techniques to a depth at least equal to the thickness of the bracket being joined. The area examined is the area of width equal to at least half the thickness of the part joined. The required stainless steel weld-deposited cladding is similarly examined. The full penetration welds are liquid-penetrant examined. Cladding thickness is required to be at least 3.2 mm. These requirements have been successfully applied to a variety of bracket designs which are attached to weld-deposited stainless steel cladding or weld buildups in many operating BWR reactor pressure vessels.

5.3.1.6.5 Time and Number of Dosimetry Measurements

GE provides a separate neutron dosimeter so that fluence measurements may be made at the vessel ID during the first fuel cycle to verify the predicted fluence at an early date in plant operation. This measurement is made over this short period to avoid saturation of the dosimeters now available. Once the fluence-to-thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between fluence and power output. It will be possible, however, to install a new dosimeter, if required, during succeeding fuel cycles.

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure head (flange) is fastened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a threaded hole in the vessel shell flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by sequential tensioning using hydraulic tensioners.

Hardness tests are performed on all main closure bolting to demonstrate that heat treatment has been properly performed.

5.3.1.8 Regulatory Guide 1.65

Regulatory Guide 1.65 defines acceptable materials and testing procedures with regard to reactor vessel closure stud bolting for light-water-cooled reactors.

The design and analysis of reactor vessel bolting materials is in full compliance with ASME Code Section III, Class I, requirements. The RPV closure studs are SA-540 Grade B23 or 24 (AISI 4340). The maximum allowable ultimate tensile strength is 1172 MPa. Also, the Charpy impact test requirements of NB-2333 will be satisfied (the lowest C_V energy will be greater than the requirement of 60.8 N•m at 13°C; the lowest reported CV expansion will exceed the 0.64 mm required).

In regards to regulatory position C.2.b, the bolting materials are ultrasonically examined in accordance with ASME Code Section III, Paragraph NB-2580, after final heat treatment and prior to threading as specified. The requirements for examination according to ASME Code Section II, SA-388 and ASTM A614 are met. The procedures approved for use in practice are judged to insure comparable material quality and are considered adequate on the basis of compliance with the applicable requirements of ASME Code Paragraph NB-2580.

The straight-beam examination is performed on 100% of cylindrical surfaces and from both ends of each stud using a 19 mm maximum diameter transducer. The reference standard for the radial scan contains a 12.7 mm diameter flat-bottom hole with a depth of 10% of the thickness. The end scan standard is per ASTM A614. Surface examinations are performed on the studs and nuts after final heat treatment and threaded as specified in the guide, in accordance with ASTM A614. Any indication greater than the indication from the applicable calibration feature is unacceptable. The distance/amplitude correction curve for the straight beam end scan of main closure studs, nuts, and washers are established as follows:

> For cylinders having a length (L) to O.D. ratio or 7 or less, the distance/amplitude curve is established by a minimum of three test points along the test distance. For cylinders having length to O. D. ratios larger than 7, the minimum number of test points is four. The test points are nearly equally spaced along the test distance. One calibration hole is located at a test distance equal to L/2.

5.3.2 Pressure/Temperature Limits

5.3.2.1 Limit Curves

The pressure/temperature limit curves in Figure 5.3-1 are based on the requirements of 10CFR50 Appendix G. The pressure/temperature limits look different than SRP Section 5.3.2 because the ABWR temperature limits are based on a more recent revision of Regulatory Guide 1.99.The pressure/temperature limits consider the effects of the predicted fluence rates from 60 years of operation which is equivalent to 52.2 Effective Full Power Years (EFPY) of reactor operation.

All the vessel shell and head areas remote from discontinuities plus the feedwater nozzles are evaluated, and the operating limit curves are based on the limiting location. The boltup limits for the flange and adjacent shell region are based on a minimum metal temperature of RT_{NDT} plus 33°C. The maximum throughwall temperature gradient from continuous heating or cooling at 55.5°C per hour was considered. The safety factors applied were as specified in ASME Code Appendix G and Reference 5.3-2.

The material for the vessel will be provided with the following requirements of RT_{NDT} as determined in accordance with Branch Technical Position MTEB 5-2: shell and flanges –20°C; nozzles –20°C and welds –20°C. Plant-specific calculations of RT_{NDT}, shift in RT_{NDT}, stress intensity factors, and pressure-temperature curves similar to those in Regulatory Guide 1.99 and SRP Section 5.3.2 will be provided in the FSAR.

5.3.2.1.1 Temperature Limits for Boltup

Minimum closure flange and fastener temperatures of RT_{NDT} plus 33°C are required for tensioning at preload condition and during detensioning. Thus, the minimum limit is –20°C + 33°C = + 13°C.

5.3.2.1.2 Temperature Limits for ISI Hydrostatic and Leak Pressure Tests

Pressure (measured in the top head) versus temperature (minimum vessel shell and head metal temperature) limits to be observed for the test and operating conditions are specified in Figure 5.3-1. Temperature limits for preservice and inservice tests are shown in Curve A of Figure 5.3-1.

5.3.2.1.3 Operating Limits During Heatup, Cooldown, and Core Operation

Heatup and Cooldown

Curve B in Figure 5.3-1 specifies limits for non-nuclear heatup and cooldown following a nuclear shutdown.

Reactor Operation

Curve C in Figure 5.3-1 specifies limits applicable for operation whenever the core is critical except for low-level physics tests.

5.3.2.1.4 Reactor Vessel Annealing

In-place annealing of the reactor vessel, because of radiation embrittlement, is not anticipated to be necessary.

5.3.2.1.5 Predicted Shift in RT_{NDT} and Drop in Upper-Shelf Energy (Appendix G-IV B)

For design purposes, the adjusted reference nil ductility temperature and drop in the upper-shelf energy for BWR vessels is predicted using the procedures in Regulatory Guide 1.99.

The calculations are based on the limits of phosphorous (0.020%), copper (0.08%) and nickel (1.2%) in the weld material. In plate material, the limits are copper (0.05%) and nickel (0.73%). Forgings will have the same chemistry as plate but the nickel limit is 1%.

An evaluation of fast neutron fluence for the Lungmen ABWR vessel was done using the Oak Ridge National Laboratory code DOT-4 on a CRAY X-MP Super Computer using an eighth core symmetry fixed source model. The neutron source was based upon a three dimensional nodal fuel model of ABWR for an integrated equilibrium core with a 26 group neutron spectrum. The results shown in Table 5.3-1 are reasonable in comparison to the BWR/6 calculations which were performed with an older version of DOT. In this comparison, the BWR/6 40 year quarter thickness evaluations for the 218-624 plant were compared to the 40 year BWR/6 238-748 plant and the 40 year Lungmen ABWR values which are shown on line three of Table 5.3-1. In evaluating the relative fluence, the power level and shroud to vessel water thickness were taken into account. In the case of the water thickness, the neutron reduction factor was interpolated from Figure 5.3-3 which shows the calculated fast neutron flux for an annular region as a function of water thickness. The incorporation of internal pumps increased the annulus between the shroud and the vessel wall for Lungmen ABWR. This leads to an order of magnitude reduction in the expected fast fluence.

A surveillance program in accordance with 10CFR50 Appendix H will be used. The surveillance program will include samples of base metal and weld metal and HAZ material, if required (see Subsection 5.3.1.6 for details on the surveillance program).

5.3.2.2 Operating Procedures

A comparison of the pressure versus temperature limit in Subsection 5.3.2.1 with intended normal operation procedures of the most severe service Level B transient shows that those limits will not be exceeded during any foreseeable upset condition. Reactor operating procedures have been established so that actual transients will not be more severe than those for which the vessel design adequacy has been demonstrated. Of the design transients, the service Level B condition producing the most adverse temperature and pressure condition anywhere in the vessel head and/or shell areas yields a minimum fluid temperature of 276°C and a maximum peak pressure of 8.38 MPaG. Scram automatically occurs as a result of this event prior to a possible reduction in fluid temperature to 121°C at a pressure of 6.41 MPaG. Per Figure 5.3-1, both the 8.38 MPaG vessel pressure at 276°C (Curve C) and the 6.41 MPaG at 121°C (Curve B) are within the calculated margin against nonductile failure.

5.3.3 Reactor Vessel Integrity

The reactor vessel material, equipment, and services associated with the reactor vessels and appurtenances would conform to the requirements of the subject purchase documents. Measures to ensure conformance included provisions for source evaluation and selection, objective evidence of quality furnished, inspection at the vendor source and examination of the completed reactor vessels.

GE provides inspection surveillance of the reactor vessel fabricator in-process manufacturing, fabrication, and testing operations in accordance with the GE quality assurance program and approved inspection procedures. The reactor vessel fabricator is responsible for the first level inspection of manufacturing, fabrication, and testing activities, and GE is responsible for the first level of audit and surveillance inspection.

Adequate documentary evidence that the reactor vessel material, manufacture, testing, and inspection conforms to the specified quality assurance requirements contained in the procurement specification is available at the fabricator plant site.

Regulatory Guide 1.2, Thermal Shock to Reactor Pressure Vessels, states that potential RPV brittle fracture, which may result from ECCS operation, need not be reviewed in individual cases if no significant changes in presently approved core and pressure vessel designs are proposed. If the margin of safety against RPV brittle fracture due to ECCS operation is considered unacceptable, an engineering solution, such as annealing, could be applied to assure adequate recovery of the fracture toughness properties of the vessel material. Regulatory Guide 1.2 requires that engineering solutions be outlined and requires demonstration that the design does not preclude use of the solutions.

An investigation of the structural integrity of BWR pressure vessels during a design basis accident (DBA) has been conducted (Reference 5.3-1). It has been determined, based on methods of fracture mechanics, that no failure of the vessel by brittle fracture as a result of DBA will occur.

The investigation included:

(1) A comprehensive thermal analysis considering the effect of blowdown and the Low-Pressure Coolant Injection System reflooding.

(2) A stress analysis considering the effects of pressure, temperature, seismic load, jet load, dead weight, and residual stresses.

(3) The radiation effect on material toughness (RT_{NDT} shift and critical stress intensity).

(4) Methods for calculating crack tip stress intensity associated with a nonuniform stress field following the design basis accident.

This analysis incorporated very conservative assumptions in all areas (particularly in the areas of heat transfer, stress analysis, effects of radiation on material toughness, and crack tip stress intensity). Therefore, because the results reported (Reference 5.3-1) provide an upper-bound approach, and it is concluded that catastrophic failure of the pressure vessel due to DBA is impossible from a fracture mechanics point of view. In the case studies, even if an acute flaw does form on the vessel inner wall, it will not propagate as the result of the DBA.

The criteria of 10CFR50 Appendix G are interpreted as establishing the requirements of annealing. Paragraph IV B requires the vessels to be designed for annealing of the beltline only where the predicted value of adjusted RT_{NDT} exceeds 93°C, as defined in Paragraph NB-2331 of ASME Code Section III. This predicted value is not exceeded; therefore, design for annealing is not required (see Subsection 5.3.1.5 for further discussion of fracture toughness of the reactor pressure vessel).

5.3.3.1 Design

5.3.3.1.1 Description

5.3.3.1.1.1 Reactor Vessel

The reactor vessel (Figures 5.3-2a and 5.3-2b and Table 5.3-2) is a vertical, cylindrical pressure vessel of welded construction. The vessel is designed, fabricated, tested, inspected, and stamped in accordance with ASME Code Section III Class 1 requirements, including the addenda in effect at the date of order placement (Table 1.8-21).

Design of the reactor vessel and its support system meets Seismic Category I equipment requirements. The materials used in the reactor pressure vessel are listed in Table 5.2-4.

The cylindrical shells, top and bottom heads and main closure flanges of the reactor vessel are fabricated of low-alloy steel, the interior of which is clad with stainless steel weld overlay except for the top head, all nozzles but the steam outlet nozzles and the reactor internal pump penetration nozzles. The bottom head is clad with Ni-Cr-Fe alloy. The reactor internal pump casings are clad with Ni-Cr-Fe alloy, or alternatively stainless steel.

In-place annealing of the reactor vessel is not necessary because shifts in transition temperature caused by irradiation during the 60-year life can be accommodated by raising the minimum pressurization temperature, and the predicted value of adjusted reference temperature does not exceed 93°C. Radiation embrittlement is not a problem outside of the vessel beltline region because the irradiation in those areas is less than $1x10^{18}$ neutron/cm^2 with neutron energies in excess of $1.6x10^{-13}$J. The use of existing methods of predicting embrittlement and operating limits which are based on a 40-year life are considered to be applicable to a 60-year life because the age degrading mechanism is irradiation and fatigue duty which are calculated for the 60-year life. Time/temperature effects will either not have any effect or will produce a small beneficial co-annealing.

Quality control methods used during the fabrication and assembly of the reactor vessel and appurtenances assure that design specifications are met.

The vessel top head is secured to the reactor vessel by studs and nuts. These nuts are tightened with a stud tensioner. The vessel flanges are sealed with two concentric metal seal-rings designed to permit no detectable leakage through the inner or outer seal at any operating condition, including heating to operating pressure and temperature at a maximum rate of 55°C in any one-hour period. To detect seal failure, a vent tap is located between the two seal rings. A monitor line is attached to the tap to provide an indication of leakage from the inner seal-ring seal.

5.3.3.1.1.2 Shroud Support

The shroud support is a circular plate welded to the vessel wall and to a cylinder supported by vertical stilt legs from the bottom head. This support is designed to carry the weight of peripheral fuel assemblies, neutron sources, core plate, top guide, shroud, shroud head and the steam separators and to laterally support the fuel assemblies and the pump diffusers. Design of the shroud support also accounts for pressure differentials across the shroud support plate, for the restraining effect of components attached to the support, and for earthquake loadings. The shroud support design is specified to meet appropriate ASME Code stress limits.

5.3.3.1.1.3 Protection of Closure Studs

The BWRs do not use borated water for reactivity control during normal operation. This subsection is therefore not applicable.

5.3.3.1.2 Safety Design Basis

The design of the reactor vessel and appurtenances meets the following safety design bases:

(1) The reactor vessel and appurtenances will withstand adverse combinations of loading and forces resulting from operation under abnormal and accident conditions.

(2) To minimize the possibility of brittle fracture of the nuclear system process barrier, the following are required:

   (a) Impact properties at temperatures related to vessel operation have been specified for materials used in the reactor vessel.

   (b) Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design and operational limitations assure that RT_{NDT} temperature shifts are accounted for in reactor operation.

   (c) Operational margins to be observed with regard to the transition temperature are specified for each mode of operation.

5.3.3.1.3 Power Generation Design Bases

The design of the reactor vessel and appurtenances meets the following power generation design bases:

(1) The reactor vessel has been designed for a useful life of 60 years.

(2) External and internal supports that are integral parts of the reactor vessel are located and designed so that stresses in the vessel and supports that result from reactions at these supports are within ASME Code limits.

(3) Design of the reactor vessel and appurtenances allows for a suitable program of inspection and surveillance.

5.3.3.1.4 Reactor Vessel Design Data

The reactor vessel design pressure is 8.62 MPaG and the design temperature is 302°C. The maximum installed test pressure is 10.78 MPaG.

5.3.3.1.4.1 Vessel Support Skirt

The vessel support skirt is constructed as an integral part of the RPV. Steel anchor bolts extend from the RPV pedestal through the flange of the skirt to secure the support skirt with the pedestal. The design is in accordance with ASME Code Section III, Division 1, NF. The connection is a friction-type joint where the bolts are pretensioned to the extent necessary to ensure that there will be no relative movement between the RPV and its pedestal. Shear forces are resisted by friction between the skirt flangeplate and the pedestal mounting plate or shear between the flange and mounting bolts.

Loading conditions are given in Table 3.9-2 of Subsection 3.9.

5.3.3.1.4.2 Control Rod Drive Housings

The control rod drive (CRD) housings are inserted through the CRD housing penetrations in the reactor vessel bottom head and are welded to Inconel stub tubes. Each housing transmits loads through the stub tubes to the bottom head of the reactor. These loads include the weights of a control rod, a control rod drive, a control rod guide tube, a four-lobed fuel-support piece, and the four fuel assemblies that rest on the fuel support piece. The housings are provided with lateral supports and are fabricated of Type-316 austenitic stainless steel.

5.3.3.1.4.3 Incore Neutron Flux Monitor Housings

Each incore neutron flux monitor housing is inserted through the incore penetrations in the bottom head, welded to Inconel stub tubes and provided with lateral supports.

An incore flux monitor guide tube is welded to the top of each housing and a startup range neutron monitor (SRNM) or a local power range monitor (LPRM) is supported from the seal/ring flange bolted at the bottom of the housing outside the vessel (Section 7.6).

5.3.3.1.4.4 Reactor Vessel Insulation

The RPV insulation is reflective metal type, constructed entirely of series 300 stainless steel and designed for a 60-year life. The insulation is made of prefabricated units engineered to fit together and maintain the insulation efficiency during temperature changes. The insulation is designed to remain in place and resist damage during a safe shutdown earthquake. Each unit is designed to permit free drainage of any moisture that may accumulate in the unit and prevent internal pressure buildup due to trapped gases.

The insulation for the RPV is supported from the biological shield wall surrounding the vessel and not from the vessel shell. Insulation for the upper head and flange is supported by a steel frame independent of the vessel and piping. During refueling, the support frame along the top head insulation is removed. The support frame is designed as a Seismic Category I structure. Insulation access panels and insulation around penetrations is designed in sections with quick release latches, which provide for ease of installation and removal for vessel inservice inspection and maintenance operation. Each insulation unit has lifting fittings attached to facilitate removal. Insulation units attached to the shield wall are not required to be readily removable except around penetrations.

At operating conditions, the insulation on the shield wall and around the refueling bellows has an average maximum heat transfer rate of 736.9 kJ/m^2/h of outside insulation surface. The maximum heat transfer rate for insulation on the top head is 682.4 kJ/m^2/h. The outside temperature of the reactor vessel is assumed to be the same as the reactor operating temperature 288°C, with the drywell air temperature being 57°C maximum. The maximum air temperature is 66°C, except for the head area above the bulkhead and refueling seal which has a maximum allowable temperature of 93°C.

5.3.3.1.4.5 Reactor Vessel Nozzles

All piping connected to the reactor vessel nozzles has been designed not to exceed the allowable loads on any nozzle. The vessel top head nozzle is provided with flanges with small groove facings. For prototype reactor internals testing, a flanged top head nozzle is provided to bolt with the flange associated with the test instrumentation. The drain nozzle is of the full penetration weld design. The feedwater inlet nozzles, core flooder inlet nozzles, and ECCS flooding nozzles have welded thermal sleeves. Nozzles connecting to stainless steel piping have safe ends or extensions made of stainless steel. The safe ends of extensions are welded to the nozzles with a SSC resistant material, Alloy 82, and welded crevices are eliminated. The welding is performed after heat treatment of the pressure vessel to avoid sensitization of the stainless steel. The material used in compiatble with the material of the mating pipe. In addition, the feedwater nozzles will be unclad and have a welded double theraml sleeve design to minimize the potential for cracking.

5.3.3.1.4.6 Materials and Inspections

The reactor vessel was designed and fabricated in accordance with the applicable ASME Boiler and Pressure Vessel Code (B&PV) as defined in Subsection 5.2.1. Table 5.2-4 defines the materials and specifications. Subsection 5.3.1.6 defines the compliance with reactor vessel material surveillance program requirements.

5.3.3.1.4.7 Reactor Vessel Schematic

The reactor vessel schematic is shown in Figure 5.3-2a.

5.3.3.2 Materials of Construction

All material used in the construction of the RPV conforms to the requirements of ASME Code Section II materials. The vessel heads, shells, flanges, and nozzles are fabricated from low-alloy steel plate and forgings purchased in accordance with ASME Specifications SA-533 Type B, Class 1 and SA-508 Class 3. Interior surfaces of the vessel are clad with austenitic stainless steel or Ni-Cr-Fe weld overlay. The material in the beltline region and below is SA-508 Class 3 forged rings.

These materials of construction were selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long term successful operating experience in reactor service.

The expected peak neutron fluence at the 0.25 t location used for evaluation is less than 6 x 10^{17} neutron/cm^2 for 60 years, the calculated shift in RT_{NDT} is 15.5°C for weld metal and 4.4°C for base metal and the drop in upper shelf energy is 13.53 N•m for welds and 10.79 N•m for base metal.

5.3.3.3 Fabrication Methods

The reactor pressure vessel is a vertical cylindrical pressure vessel of welded construction fabricated in accordance with ASME Code Section III, Class 1, requirements. All fabrication of the reactor pressure vessel was performed in accordance with GE-approved drawings, fabrication procedures, and test procedures. The shell and vessel head were made from formed low-alloy steel plates or forgings and the flanges and nozzles from low-alloy steel forgings. Welding performed to join these vessel components was in accordance with procedures qualified to ASME Section III and IX requirements. Weld test samples were required for each procedure for major-vessel full-penetration welds.

Submerged arc and manual stick electrode welding processes were employed. Electroslag welding was not applied. Preheat and interpass temperatures employed for welding of low-alloy steel met or exceeded the requirements of ASME Section III, Appendix D. Post-weld heat treatment of 593°C minimum was applied to all low-alloy steel welds.

All previous BWR pressure vessels have employed similar fabrication methods. These vessels have operated for an extensive number of years and their service history is rated excellent.

5.3.3.4 Inspection Requirements

All plates, forgings, and bolting are 100% ultrasonically tested and surface examined by magnetic-particle methods or liquid-penetrant methods in accordance with ASME Code Section III. Welds on the reactor pressure vessel are examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME Code Section III. In addition, the pressure-retaining welds are ultrasonically examined using acceptance standards which are required by ASME Code Section XI.

5.3.3.5 Shipment and Installation

The completed reactor vessel is given a thorough cleaning and examination prior to shipment. The vessel is tightly sealed for shipment to prevent entry of dirt or moisture. Preparations for shipment are in accordance with detailed written procedures.

On arrival at the reactor site, the reactor vessel is examined for evidence of any contamination as a result of damage to shipping covers. Measures are taken during installation to assure that vessel integrity is maintained; for example, access controls are applied to personnel entering the vessel, weather protection is provided, and periodic cleanings are performed.

5.3.3.6 Operating Conditions

Procedural controls on plant operation are implemented to hold thermal stresses within acceptable ranges and to meet the pressure/temperature limits of Subsection 5.3.2. The restrictions on coolant temperature are as follows:

(1) The average rate of change of reactor coolant temperature during normal heatup and cooldown shall not exceed 55°C during any one-hour period.

(2) If the coolant temperature difference between the dome (inferred from Psat) and the bottom head drain exceeds 80°C, neither reactor power level nor recirculation pump flow shall be increased.

The limit regarding the normal rate of heatup and cooldown (Item 1) assures that the vessel closure, closure studs, vessel support skirt, Control Rod Drive System (CRD) housing, and stub tube stresses and usage remain within acceptable limits. Vessel temperature limit on recirculating pump operation and power level increase restriction (Item 2) augments the Item 1 limit in further detail by assuring that the vessel bottom head region will not be warmed at an excessive rate caused by rapid sweep out of cold coolant in the vessel lower head region by recirculation pump operation or natural circulation (cold coolant can accumulate as a result of control drive inleakage and/or low recirculation flow rate during startup or hot standby).

These operational limits, when maintained, ensure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel was designed for normal operating conditions. To maintain the integrity of the vessel in the event that these operational limits are exceeded, the reactor vessel has been designed to withstand a limited number of transients caused by operator error. Also, for abnormal operating conditions where safety systems or controls provide an automatic temperature and pressure response in the reactor vessel, the reactor vessel integrity is maintained, since the severest anticipated transients have been included in the design conditions. Therefore, it is concluded that the vessel integrity will be maintained during the most severe postulated transients, since all such transients are evaluated in the design of the reactor vessel.

5.3.3.7 Inservice Surveillance

Inservice inspection of the RPV will be in accordance with the requirements of ASME B&PV Code Section XI. The vessel will be examined once prior to startup to satisfy the preoperational requirements of IWB-2000 of ASME Code Section XI. Subsequent inservice inspection will monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and thermal environment. Specimens of actual reactor beltline material will be exposed in the reactor vessel and periodically withdrawn for impact testing. Operating procedures will be modified in accordance with test results to assure adequate brittle-fracture control.

Material surveillance programs and inservice inspection programs are in accordance with applicable ASME Code requirements and provide assurance that brittle-fracture control and pressure vessel integrity will be maintained throughout the service lifetime of the reactor pressure vessel.

5.3.4 References

5.3-1 *An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident,* NEDO-10029, June 1969.

5.3-2 *Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors,* NEDO-21778-A, January 1979.

Table 5.3-1 Comparison of 40 Year Fluences

	BWR/6		ABWR
	218-624	238-748	
Peak Fluence (40y) (0.25t)	5.5E+18	4.3E+18	2.2E+17
Power (MWt)	2894	3579	3926
Bundles	624	748	872
Power Density (kW/L)	52.8	54.5	51.3
Vessel IR (cm)	276.86	302.26	353.06
Shroud OR (cm)	234.95	256.54	280.35
Water Gap (cm)	41.9	45.7	72.7
Neutron Reduction Factor for Water	0.007	0.0044	0.00042
Expected Fast Fluence based upon 218-624	5.5E+18	3.6E+18	3.3E+17

Table 5.3-2 Key Dimensions of RPV System Components and Acceptable Variations

Description	Dimension/ Elevation (Figure 5.3-2a)	Nominal Value (mm)	Acceptable Variation (mm)
RPV inside diameter (Inside cladding)	A	7112.0	±51.0
RPV wall thickness in beltline (without cladding)	B	178.0	+20.0/–4.0
RPV bottom head inside invert. Elevation	C	0.0	Reference
RPV support skirt flange bottom, Elevation	D	3300.0	±75.0
Core plate support/Top of shroud middle flange, Elevation	E	4695.2	±15.0
Top guide support/Top of shroud top flange, Elevation	F	9351.2	±20.0
RPV stabilizer connection, Elevation	G	13,766.0	±20.0
Top of RPV flange, Elevation	H	17,703.0	±65.0
RHR SDC/RWCU Outlet Nozzle, Elevation	J	10,921.0	±40.0
Shroud outside diameter	K	5600.7	±25.0
Shroud wall thickness	L	57.2	±10.0
Shroud support legs (Fig. 5.3-2b)	M x N	662.0 x 153.0	±20.0 for M ±10.0 for N
Control rod guide tube outside diameter	P	273.05	±5.0

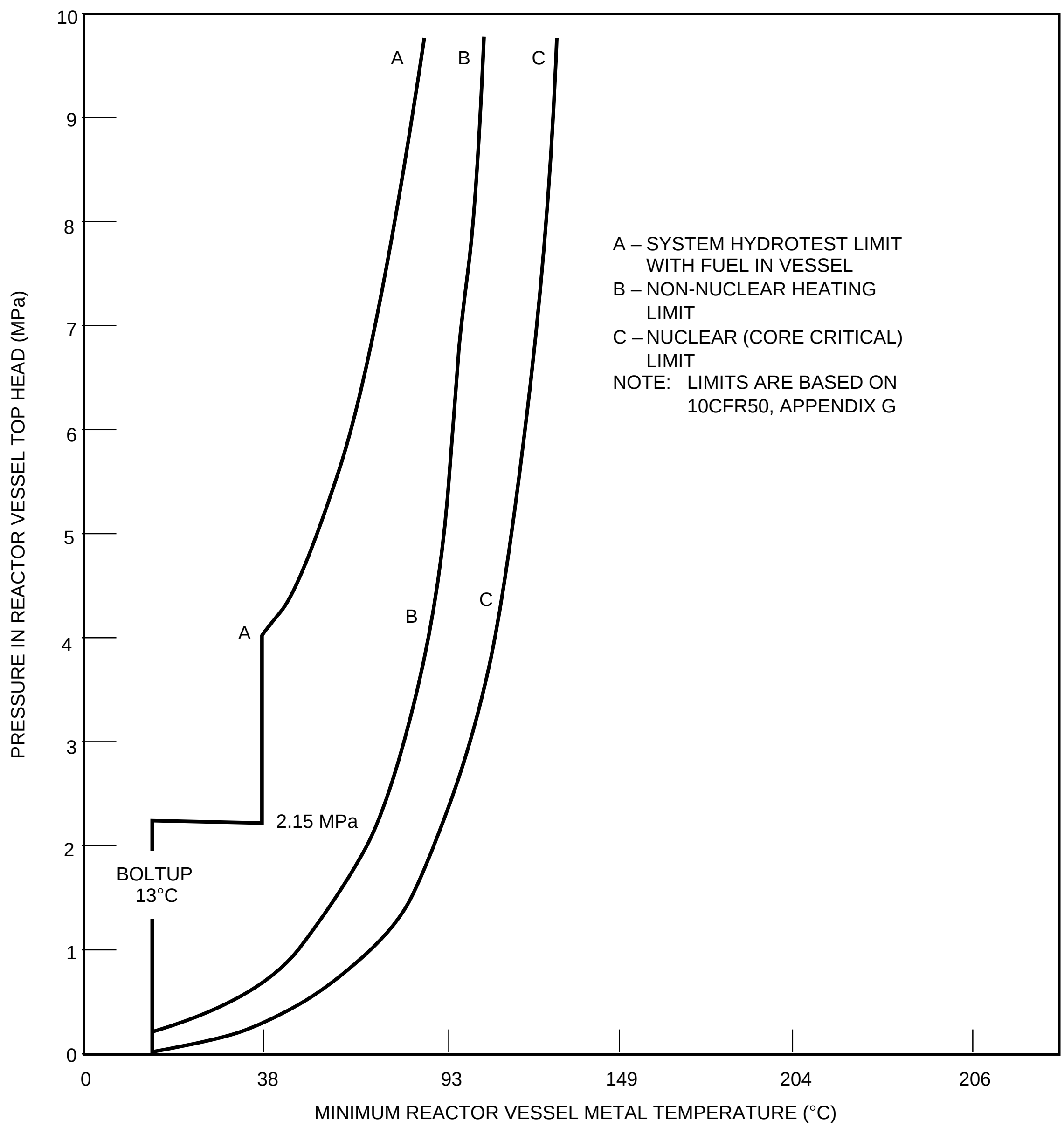

Figure 5.3-1 Minimum Temperature Required Versus Reactor Pressure

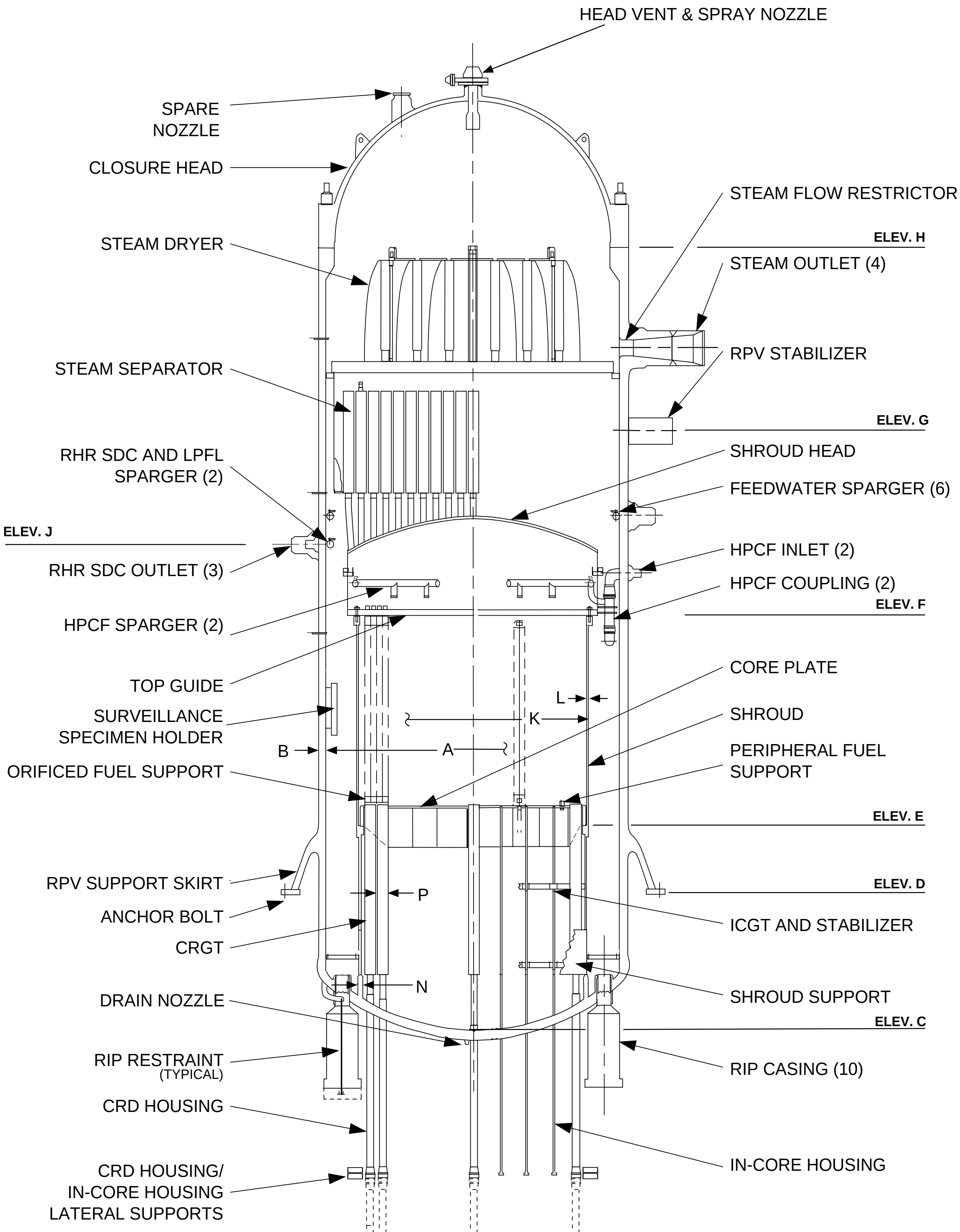

Figure 5.3-2a Reactor Pressure Vessel System (RPV) Key Features

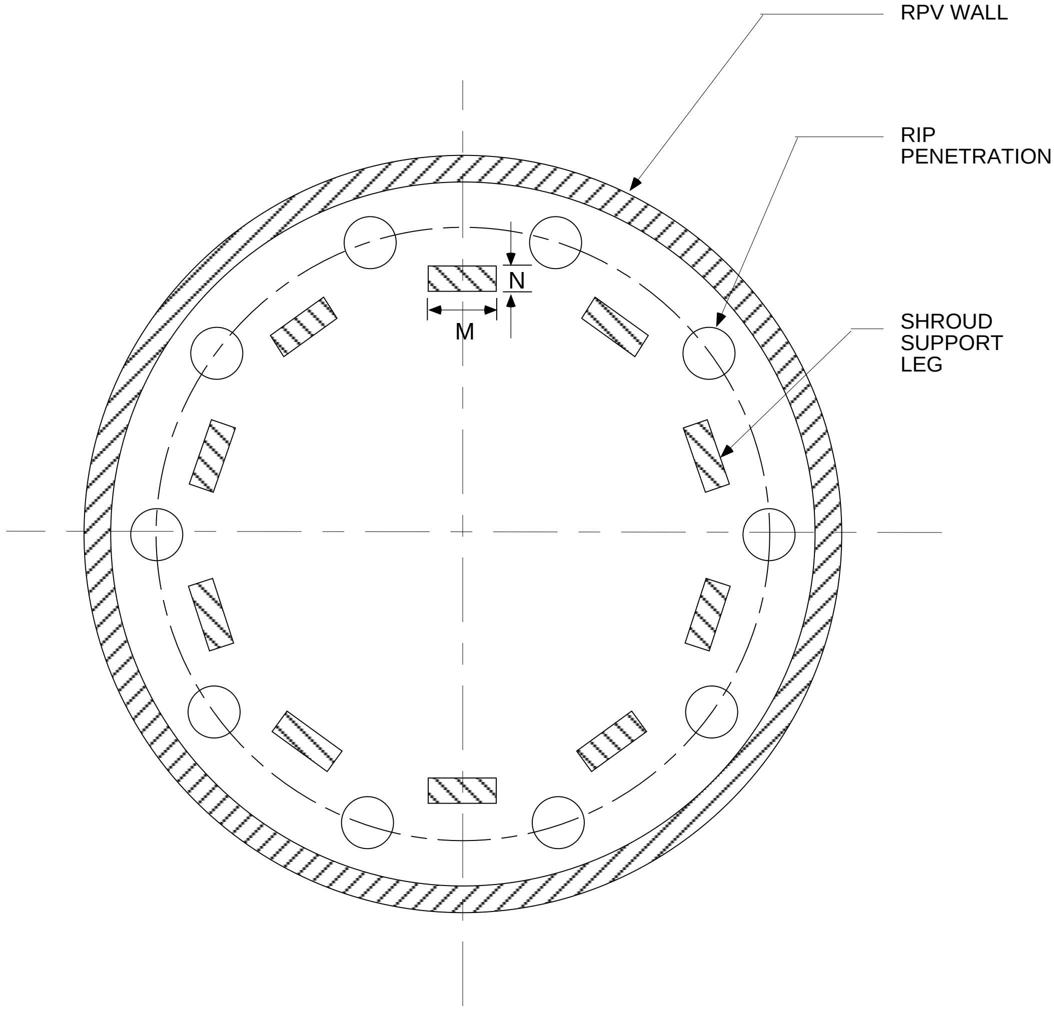

Figure 5.3-2b Pump Penetration and Shroud Leg Arrangement

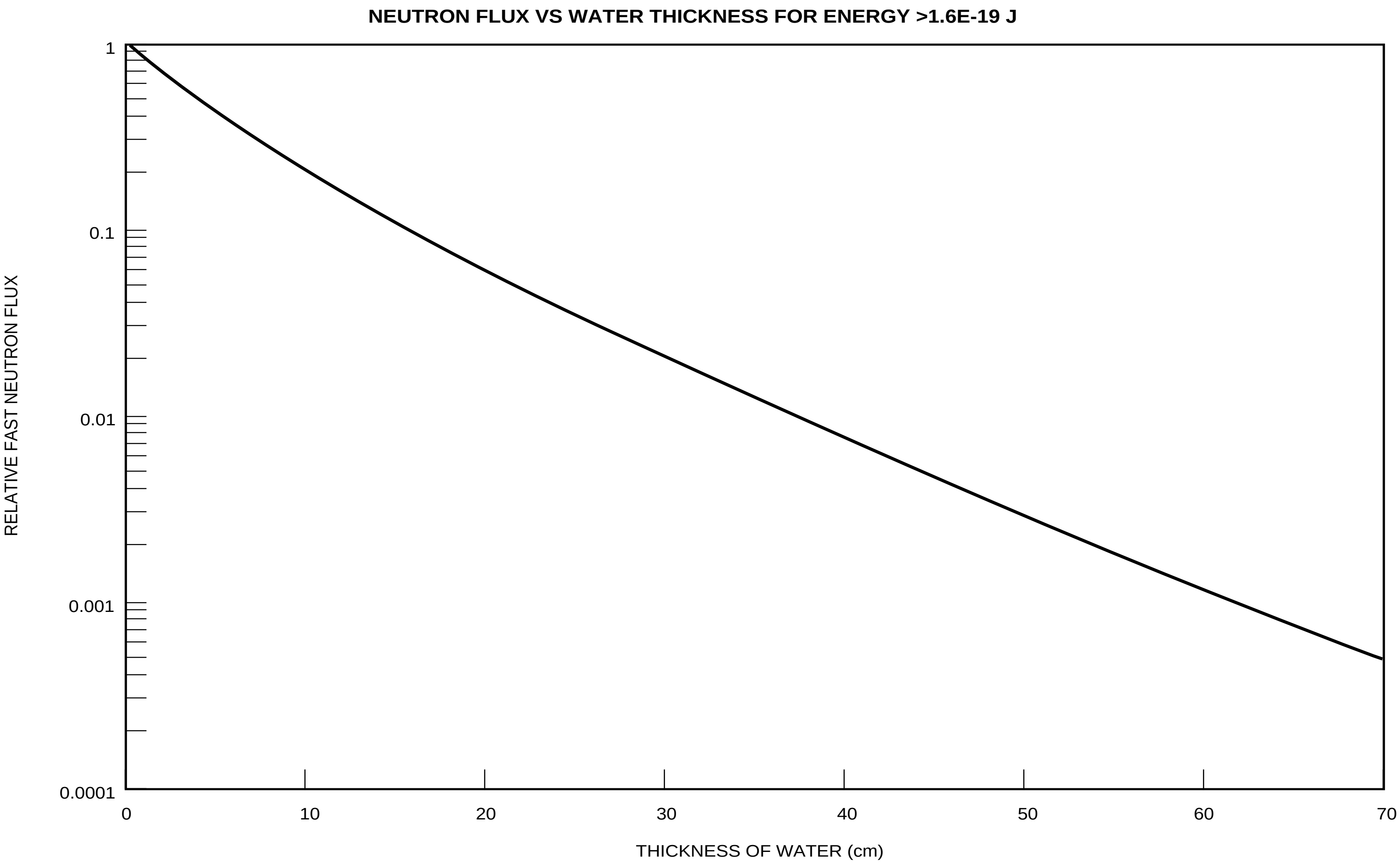

Figure 5.3-3 Fast Neutron Flux as Function of Water Thickness