5.3   Reactor Vessel

5.3.1   Reactor Vessel Design

5.3.1.1   Safety Design Bases

The reactor vessel, as an integral part of the reactor coolant pressure boundary will be designed, fabricated, erected and tested to quality standards commensurate with the requirements set forth in 10 CFR 50, 50.55a and General Design Criterion 1. Design and fabrication of the reactor vessel is carried out in accordance with ASME Code, Section III, Class 1 requirements. Subsections 5.2.3 and 5.3.2 provide further details.

The performance and safety design bases of the reactor vessel follow:

- The reactor vessel provides a high integrity pressure boundary to contain the reactor coolant, heat generating reactor core, and fuel fission products. The reactor vessel is the primary pressure boundary for the reactor coolant and the secondary barrier against the release of radioactive fission products.

- The reactor vessel provides support for the reactor internals, flow skirt, and core to ensure that the core remains in a coolable configuration.

- The reactor vessel directs main coolant flow through the core by close interface with the reactor internals and flow skirt.

- The reactor vessel provides for core internals location and alignment.

- The reactor vessel provides support and alignment for the control rod drive mechanisms and in-core instrumentation assemblies.

- The reactor vessel provides support and alignment for the integrated head assembly.

- The reactor vessel provides an effective seal between the refueling cavity and sump during refueling operations.

- The reactor vessel supports and locates the main coolant loop piping.

- The reactor vessel provides support for safety injection flow paths.

- The reactor vessel serves as a heat exchanger during core meltdown scenario with water on the outside surface of the vessel.

5.3.1.2   Safety Description

The reactor vessel consists of a cylindrical section with a transition ring, hemispherical bottom head, and a removable flanged hemispherical upper head (Figure 5.3-1). Key dimensions are shown in Figures 5.3-5 and 5.3-6. The cylindrical section consists of two shells, the upper shell and the lower shell. The upper and lower shells and the lower hemispherical head are fabricated

from low alloy steel and clad with austenitic stainless steel. The upper shell forging is welded to the lower shell forging, and the lower shell is welded to the transition ring, which is welded to the hemispherical bottom head. The removable flanged hemispherical upper head consists of a single forging, which includes the closure head flange and the closure head dome. The closure head is fabricated from a low alloy steel forging and clad with austenitic stainless steel. Specifics of the processes used in base materials, clad material, and weld materials are discussed in subsection 5.2.3. The removable flanged hemispherical closure head is attached to the vessel (consisting of the upper shell-lower shell-bottom hemispherical head) by studs. Two metal o-rings are used for sealing the two assemblies. Inner and outer monitor tubes are provided through the upper shell to collect any leakage past the o-rings. Details of the head gasket monitoring connections are included in subsection 5.2.5.2.1.

The reactor vessel supports the internals. An internal ledge is machined into the top of the upper shell section. The core barrel flange rests on the ledge. A large circumferential spring is positioned on the top surface of the core barrel flange. The upper support plate rests on the top surface of the spring. The spring is compressed by installation of the reactor vessel closure head and the upper and lower core support assemblies are restrained from any axial movements.

Four core support pads are located on the bottom hemispherical head just below the transition ring-to-lower shell circumferential weld. The core support pads function as a clevis. At assembly, as the lower internals are lowered into the vessel, the keys at the bottom of the lower internals engage the clevis in the axial direction. With this design, the internals are provided with a lateral support at the furthest extremity and may be viewed as a beam supported at the top and bottom.

The interfaces between the reactor vessel and the lower internals core barrel are such that the main coolant flow enters through the inlet nozzle and is directed down through the annulus between the reactor vessel and core barrel and through the flow skirt and flows up through the core. The annulus is designed such that the core remains in a coolable configuration for all design conditions.

Prior to installation of the internals into the reactor vessel, guide studs are assembled into the upper shell. Dimensional relationships are established between the guide studs and the core support pads such that when the lower internals lifting rig engages the guide studs, the keys at the bottom of the lower internals are in relative circumferential position to enter the core support pads.

There are 69 penetrations in the removable flanged hemispherical head (closure head) that are used to provide access for the control rod drive mechanisms. Each control rod drive mechanism is positioned in its opening and welded to the closure head penetration. In addition there are eight penetrations in the closure head used to provide access for in-core and core exit instrumentation.

Lugs are welded to the outside surface of the closure head along the outer periphery of the dome section. The purpose of these lugs is to provide support and alignment for the integrated head package.

Attached to the top surface and along the outer periphery of the upper shell is a ring section. During field assembly the ring is welded to the refueling cavity seal liner. This ring provides an effective water seal between the refueling cavity and sump during refueling operations.

A support pad is integral to each of the four inlet nozzles. The reactor vessel is supported by the pads. The pads rest on steel base pads atop a support structure, which is attached to the concrete foundation wall. Thermal expansion and contraction of the vessel are accommodated by sliding surfaces between the support pads and the base plates. Side stops on these plates keep the vessel centered and resist lateral loads.

The reactor vessel primary and direct vessel injection (DVI) nozzles are located in the upper shell. These nozzles are either forged as part of the upper shell forging or are fabricated by "set in" construction such that the welding is through the vessel shell forging. A stainless steel safe end is shop welded to each of the four inlet, two outlet and two DVI nozzles to facilitate field welding without heat treatment to the stainless steel reactor coolant piping system. The primary coolant nozzles support one end of the primary coolant system. Reaction loads are transferred into the nozzles and eventually into the support pads. The inlet and outlet elevation nozzles are offset in different planes by 17.5 inches. This allows pump maintenance without discharging the core.

There are no penetrations in the reactor vessel below the core. This eliminates the possibility of a loss-of-coolant accident by leakage from the reactor vessel that would allow the core to be uncovered.

5.3.1.3   System Safety Evaluation

The reactor vessel is part of the reactor coolant system. Load and stress evaluation for operating loads and mechanical transients of safe shutdown earthquake (SSE), and pipe ruptures appear in subsection 3.9.3.

5.3.1.4   Inservice Inspection/Inservice Testing

Inservice surveillance is discussed in subsection 5.3.4.7.

5.3.2   Reactor Vessel Materials

5.3.2.1   Material Specifications

Material specifications are in accordance with the ASME Code requirements and are given in subsection 5.2.3. All ferritic reactor vessel materials comply with the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR 50.

The ferritic materials of the reactor vessel beltline are restricted to the maximum limits shown in Table 5.3-1. Copper, nickel, and phosphorus content is restricted to reduce sensitivity to irradiation embrittlement in service.

5.3.2.2   Special Processes Used for Manufacturing and Fabrication

The reactor vessel is classified as AP1000 Class A. Design and fabrication of the reactor vessel is carried out in accordance with ASME Code, Section III, Class 1 requirements. The shell sections, flange, and nozzles are manufactured as forgings. The hemispherical heads are made from dished plates or forgings. The reactor vessel parts are joined by welding, using the single or multiple wire submerged arc and the shielded metal arc processes. Gas metal arc welding and plasma arc welding are acceptable methods of applying buttering for dissimilar metal welds.

The use of severely sensitized stainless steel as a pressure boundary material is prohibited and is eliminated by either a select choice of material or by programming the method of assembly.

At locations in the reactor vessel where stainless steel and nickel-chromium-iron alloy are joined, the final joining beads are nickel-chromium-iron alloy weld metal in order to prevent cracking.

The location of full penetration weld seams in the upper closure head and vessel bottom head are restricted to areas that permit accessibility during in-service inspection.

The stainless steel clad surfaces are sampled to demonstrate that composition requirements are met.

Freedom from underclad cracking is provided by special evaluation of the procedure qualification for cladding applied on low-alloy steel (SA-508, GR 3 CL 1).

Minimum preheat requirements have been established for pressure boundary welds using low-alloy material. The preheat is maintained until either a low temperature (400°F – 500°F) post heat treatment, an intermediate postweld heat treatment or a full postweld heat treatment is performed.

A field weld is made, after the reactor vessel has been set, to install the permanent reactor vessel cavity seal ring. This stainless steel filler weld joins the seal ring to the reactor vessel seal ledge. A minimum preheat is specified for this weld in compliance with the ASME Code requirements.

The flow skirt is also welded to support lugs in the field after the reactor vessel/internals system is set.

5.3.2.3   Special Methods for Nondestructive Examination

The nondestructive examination (NDE) of the reactor vessel and its appurtenances is conducted in accordance with ASME Code, Section III requirements; also, numerous examinations are performed in addition to ASME Code, Section III requirements. The nondestructive examination of the vessel is discussed in the following paragraphs, and the reactor vessel quality assurance program is given in Table 5.3-2.

5.3.2.3.1 Ultrasonic Examination

In addition to the required ASME Code straight beam ultrasonic examination, angle beam inspection over 100 percent of one major surface of plate material is performed during fabrication to detect discontinuities that may be undetected by the straight beam examination.

In addition to the ASME Code, Section III nondestructive examination, full penetration ferritic pressure boundary welds in the reactor vessel are ultrasonically examined during fabrication. This test is performed upon completion of the welding and intermediate heat treatment but prior to the final postweld heat treatment.

After hydrotesting, full penetration ferritic pressure boundary welds in the reactor vessel, as well as the nozzle to safe end welds, are ultrasonically examined. These inspections are performed in addition to the ASME Code, Section III nondestructive examination requirements.

5.3.2.3.2 Penetrant Examinations

The partial penetration welds for the control rod drive mechanism head adapters and the QuickLoc assemblies are inspected by dye penetrant after the root pass, in addition to ASME code requirements. Core support block attachment welds are inspected by dye penetrant after the first layer of weld metal and after each 0.5 inch of weld metal. Clad surfaces and other vessel and head internal surfaces are inspected by dye penetrant after the hydrostatic test.

5.3.2.3.3 Magnetic Particle Examination

Magnetic particle examination requirements below are in addition to the magnetic particle examination requirements of Section III of the ASME Code. All magnetic particle examinations of materials and welds are performed in accordance with the following:

- Prior to the final postweld heat treatment, only by the prod, coil, or direct contact method
- After the final postweld heat treatment, only by the yoke method

The following surfaces and welds are examined by magnetic particle methods. The acceptance standards are in accordance with Section III of the ASME Code.

Surface Examinations

- Magnetic particle examination of exterior vessel and head surfaces after the hydrostatic test.

- Magnetic particle examination of exterior closure stud surfaces and all nut surfaces after final machining or rolling. Continuous circular and longitudinal magnetization is used.

- Magnetic particle examination of inside diameter surfaces of carbon and low alloy steel products that have their properties enhanced by accelerated cooling. This inspection is performed after forming and machining and prior to cladding.

Weld Examination

Magnetic particle examination of the welds attaching the closure head lifting lugs and refueling seal ledge to the reactor vessel after the first layer and each 0.5 inch of weld metal is deposited. All pressure boundary welds are examined after back-chipping or back-grinding operations.

5.3.2.4   Special Controls for Ferritic and Austenitic Stainless Steels

Welding of ferritic steels and austenitic stainless steels is discussed in subsection 5.2.3. Subsection 5.2.3 includes discussions on the degree of conformance with Regulatory Guide 1.44. Section 1.9 discusses the degree of conformance with Regulatory Guides, including 1.31 and 1.34 (if applicable), as well as 1.37, 1.43, 1.50, 1.71, and 1.99.

5.3.2.5   Fracture Toughness

Assurance of adequate fracture toughness of ferritic materials in the reactor vessel (ASME Code, Section III, Class 1 component) is provided by compliance with the requirements for fracture toughness testing included in NB-2300 to Section III of the ASME Code and Appendix G of 10 CFR 50.

The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline base metal transverse direction and welds are 75 foot-pounds, as required by Appendix G of 10 CFR 50. The vessel fracture toughness data are given in Table 5.3-3. The AP1000 end-of-life RT_{NDT} and upper shelf energy projections were estimated using Regulatory Guide 1.99 for the end-of-life neutron fluence at the 1/4-thickness (T) and ID reactor vessel locations.

5.3.2.6   Material Surveillance

In the surveillance program, the evaluation of radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and postirradiation testing of Charpy V-notch, tensile, and 1/2-T compact tension (CT) fracture mechanics test specimens. The program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach. The program conforms to ASTM E-185 (Reference 1) and 10 CFR 50, Appendix H.

The reactor vessel surveillance program incorporates eight specimen capsules. The capsules are located in guide baskets welded to the outside of the core barrel as shown in Figure 5.3-4 and positioned directly opposite the center portion of the core. The capsules can be removed when the vessel head is removed. To meet the guidelines of ASTM E-185 (lead factors less than three), the specimen guide baskets are located azimuthally near the lowest fluence locations at 135, 225, and 315 degrees. The 45 degree location is also a low fluence azimuthal location; however, there is a Roto-Lock insert for the internals lifting rig, which would prevent access for removal of the capsules from the baskets. Therefore, there are no guide baskets at the 45 degree location. Eight specimen capsules are provided by including three guide baskets at the 135 and 315 degree azimuthal locations and two baskets at the 225 degree location.

The capsules contain reactor vessel weld metal, base metal, and heat-affected zone metal specimens. The base metal specimens are oriented both parallel and normal (longitudinal and

transverse) to the principal rolling direction of the limiting base material located in the core region of the reactor vessel. The 8 capsules contain 72 tensile specimens, 480 Charpy V-notch specimens, and 48 compact tension specimens. Archive material sufficient for two additional capsules and heat-affected-zone (HAZ) materials is retained.

Dosimeters, as described below, are placed in filler blocks drilled to contain them. The dosimeters permit evaluation of the flux seen by the specimens and the vessel wall. In addition, thermal monitors made of low melting point alloys are included to monitor the maximum temperature of the specimens. The specimens are enclosed in a tight-fitting stainless steel sheath to prevent corrosion and ensure good thermal conductivity. The complete capsule is helium leak tested. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01 percent is made for surveillance material and as deposited weld metal. Each of the eight capsules contains the specimens shown in Table 5.3-4.

The following dosimeters and thermal monitors are included in each of the eight capsules:

- Dosimeters

  - Iron
  - Copper
  - Nickel
  - Niobium-93 (cadmium shielded)
  - Cobalt-aluminum (0.15-percent cobalt)
  - Cobalt-aluminum (cadmium shielded)

- Thermal Monitors

  - 97.5-percent lead, 2.5-percent silver, (579°F melting point)
  - 97.5-percent lead, 1.75-percent silver, 0.75-percent tin (590°F melting point)

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the vessel wall, with the specimens being located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the transition temperature shift measurements are representative of the vessel at a later time in life. The lead factors for the eight specimen capsule locations based on the reference neutron flux distribution (flux distribution that results in the maximum fluence on the reactor vessel inner surface) vary between approximately 1.8 and 2.3. These lead factors will change over the life of the plant due to changes in core design and operating parameters. Data from CT fracture toughness specimens are expected to provide additional information for use in determining allowable stresses for irradiated material.

Correlations between the calculations and measurements of the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in subsection 5.3.2.6.1. The anticipated degree to which the specimens perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure is made by the use of data on capsules withdrawn. The recommended program schedule for removal of the capsules for

post-irradiation testing includes five capsules to be withdrawn instead of four as specified in ASTM E-185 (Reference 1) and Appendix H of 10 CFR 50. The following is the recommended withdrawal schedule of capsules for AP1000.

Capsule	Withdrawal Time
1st	When the accumulated neutron fluence of the capsule is 5×10^{18} n/cm^2.
2nd	When the accumulated neutron fluence of the capsule corresponds to the approximate end of life fluence at the reactor vessel 1/4T location.
3rd	When the accumulated neutron fluence of the capsule corresponds to the approximate end of life fluence at the reactor vessel inner wall location.
4th	When the accumulated neutron fluence of the capsule corresponds to a fluence not less than once or greater than twice the peak end of vessel life fluence.
5th	End of plant design objective of 60 years
6th	Standby
7th	Standby
8th	Standby

5.3.2.6.1 Measurement of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

The use of passive neutron sensors such as those included in the internal surveillance capsule dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period. An accurate estimate of the average neutron flux level, and hence, time integrated exposure (fluence) experienced by the sensors may be derived from the activation measurements only if the parameters of the irradiation are well known. In particular, the following variables are of interest:

- The measured specific activity of each sensor
- The physical characteristics of each sensor
- The operating history of the reactor
- The energy response of each sensor
- The neutron energy spectrum at the sensor location

The procedures used to determine sensor specific activities, to develop reaction rates for individual sensors from the measured specific activities and the operating history of the reactor, and to derive key fast neutron exposure parameters from the measured reaction rates are described below.

5.3.2.6.1.1 Determination of Sensor Reaction Rates

The specific activity of each of the radiometric sensors is determined using established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor is determined by means of a high purity germanium gamma spectrometer. In the case of the surveillance capsule multiple foil sensor sets, these analyses are performed by direct counting of each of the individual wires or, as in the case of niobium monitors, by appropriate methods as described in ASTM E 1297.

The irradiation history of the reactor over its operating lifetime is determined from plant power generation records. In particular, operating data are extracted on a monthly basis from reactor startup to the end of the capsule irradiation period. For the sensor sets utilized in the surveillance capsule irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation are determined from the following equation:

$$R = \frac{A}{N_0 \, F \, Y \, \sum_j \dfrac{P_j}{P_{ref}} C_j [1 - e^{-\lambda t_j}] e^{-\lambda t_d}}$$

where:

A = measured specific activity provided in terms of disintegrations per second per gram of target material (dps/gm).

R = reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} expressed in terms of reactions per second per nucleus of target isotope (rps/nucleus).

N_0 = number of target element atoms per gram of sensor.

F = weight fraction of the target isotope in the sensor material.

Y = number of product atoms produced per reaction.

P_j = average core power level during irradiation period j (MW).

P_{ref} = maximum or reference core power level of the reactor (MW).

C_j = calculated ratio of $\phi(E > 1.0 \text{ MeV})$ during irradiation period j to the time weighted average $\phi(E > 1.0 \text{ MeV})$ over the entire irradiation period.

λ = decay constant of the product isotope (sec-1).

t_j =   length of irradiation period j (sec).

t_d =   decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the total irradiation period.

In the above equation, the ratio P_j/P_{ref} accounts for month-by-month variation of power level within a given fuel cycle. The ratio C_j is calculated for each fuel cycle and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. Since the neutron flux at the measurement locations within the surveillance capsules is dominated by neutrons produced in the peripheral fuel assemblies, the change in the relative power in these assemblies from fuel cycle to fuel cycle can have a significant impact on the activation of neutron sensors. For a single-cycle irradiation, $C_j = 1.0$. However, for multiple-cycle irradiations, particularly those employing low leakage fuel management, the additional C_j correction must be utilized in order to provide accurate determinations of the decay corrected reaction rates for the dosimeter sets contained in the surveillance capsules.

5.3.2.6.1.2 Least Squares Adjustment Procedure

Least squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as neutron fluence (E > 1.0 MeV) or iron atom displacements (dpa) along with their uncertainties are then easily obtained from the adjusted spectrum. The use of measurements in combination with the analytical results reduces the uncertainty in the calculated spectrum and acts to remove biases that may be present in the analytical technique.

In general, the least squares methods, as applied to pressure vessel fluence evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, Ri, to a single neutron spectrum, ϕg, through the multigroup dosimeter reaction cross-section, σig, each with an uncertainty δ.

The use of least squares adjustment methods in LWR dosimetry evaluations is not new. The American Society for Testing and Materials (ASTM) has addressed the use of adjustment codes in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance" and many industry workshops have been held to discuss the various applications. For example, the ASTM-EURATOM Symposia on Reactor Dosimetry holds workshops on neutron spectrum unfolding and adjustment techniques at each of its bi-annual conferences.

The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement. The analytical method alone may be deficient because it inherently contains uncertainty due to the input assumptions to the calculation. Typically these assumptions include parameters such as the temperature of the water in the peripheral fuel assemblies, by-pass region, and downcomer regions, component dimensions, and peripheral core source. Industry consensus indicates that the use of calculation alone results in overall uncertainties in the neutron exposure parameters in the range of 15-20% (1σ).

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For a given application, the calculated neutron spectrum is obtained from the results of plant specific neutron transport calculations applicable to the irradiation period experienced by the dosimetry sensor set. This calculation is performed using the benchmarked transport calculational methodology described in Section 5.3.2.6.2. The sensor reaction rates are derived from the measured specific activities obtained from the counting laboratory using the specific irradiation history of the sensor set to perform the radioactive decay corrections. The dosimetry reaction cross-sections and uncertainties that are utilized in LWR evaluations comply with ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)."

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum are input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties also follows the guidance provided in ASTM Standard E 944.

5.3.2.6.2   Calculation of Integrated Fast Neutron (E > 1.0 MeV) Flux at the Irradiation Samples

A generalized set of guidelines for performing fast neutron exposure calculations within the reactor configuration, and procedures for analyzing measured irradiation sample data that can be correlated to these calculations, has been promulgated by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.190, or RG-1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Reference 2]. Since different calculational models exist and are continuously evolving along with the associated model inputs, e.g., cross-section data, it is worthwhile summarizing the key models, inputs, and procedures that the NRC staff finds acceptable for use in determining fast neutron exposures within the reactor geometry. This material is highlighted in the subsection of material that is provided below.

5.3.2.6.2.1 Calculation and Dosimetry Measurement Procedures

The selection of a particular geometric model, the corresponding input data, and the overall methodology used to determine fast neutron exposures within the reactor geometry are based on the needs for accurately determining a solution to the problem that must be solved and the data/resources that are currently available to accomplish this task. Based on these constraints, engineering judgment is applied to each problem based on an analyst's thorough understanding of the problem, detailed knowledge of the plant, and due consideration to the strengths and weaknesses associated with a given calculational model and/or methodology. Based on these conditions, RG-1.190 does not recommend using a singular calculational technique to determine fast neutron exposures. Instead, RG-1.190 suggests that one of the following neutron transport tools be used to perform this work.

- Discrete Ordinates Transport Calculations

    – Adjoint calculations benchmarked to a reference-forward calculation, or stand-alone forward calculations.

    – Various geometrical models utilized with suitable mesh spacing in order to accurately represent the spatial distribution of the material compositions and source.

    – In performing discrete ordinates transport calculations, RG-1.190 also suggests that a P_3 angular decomposition of the scattering cross-sections be used, as a minimum.

    – RG-1.190 also recommends that discrete ordinates transport calculations utilize S_8 angular quadrature, as a minimum.

    – RG-1.190 indicates that the latest version of the Evaluated Nuclear Data File, or ENDF/B, should be used for determining the nuclear cross-sections; however, cross-sections based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked are also acceptable.

- Monte Carlo Transport Calculations

    A complete description of the Westinghouse pressure vessel neutron fluence methodology, which is based on discrete ordinates transport calculations, is provided in Reference 3. The Westinghouse methodology adheres to the guidelines set forth in Regulatory Guide 1.190.

5.3.2.6.2.2 Plant-Specific Calculations

The location, selection, and evaluation of neutron dosimetry and the associated radiometric monitors, as well as fast (E > 1.0 MeV) neutron fluence assessments of the AP1000 reactor pressure vessel, are conducted in accordance with the guidelines that are specified in Regulatory Guide 1.190.

5.3.2.7   Reactor Vessel Fasteners

The reactor vessel closure studs, nuts, and washers are designed and fabricated in accordance with the requirements of the ASME Code, Section III. The closure studs are fabricated of SA-540. The closure stud material meets the fracture toughness requirements of the ASME Code, Section III, and 10 CFR 50, Appendix G. Conformance with Regulatory Guide 1.65, Materials and Inspections for Reactor Vessel Closure Studs, is discussed in Section 1.9. Nondestructive examinations are performed in accordance with the ASME Code, Section III. See subsection 5.2.3 for restrictions on lubricants.

Refueling procedures require that the reactor vessel closure studs, nuts, and washers are lifted out of their respective holes and a stud support collar be put in place prior to the lift of the integrated head assembly during preparation for refueling. In this way the studs are lifted with and stored on the head. An alternative method is to remove the reactor vessel closure studs, nuts, and washers from the reactor closure and place them in storage racks during preparation for refueling. In this method, the storage racks are removed from the refueling cavity and stored at convenient locations on the containment operating deck prior to removal of the reactor closure head and refueling cavity flooding. In either case, the reactor closure studs are not exposed to the borated refueling cavity water. Additional protection against the possibility of incurring corrosion effects is provided by the use of a manganese base phosphate surfacing treatment.

The stud holes in the reactor flange are sealed with special plugs before removing the reactor closure, thus preventing leakage of the borated refueling water into the stud holes.

5.3.3   Pressure-Temperature Limits

5.3.3.1   Limit Curves

Heatup and cooldown pressure-temperature limit curves are required as a means of protecting the reactor vessel during startup and shut down to minimize the possibility of fast fracture. The methods outlined in Appendix G of Section III of the ASME Code are employed in the analysis of protection against nonductile failure. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil ductility temperature, which includes a reference nil ductility temperature shift (ΔRT_{NDT}), initial RT_{NDT} and margin. The extent of the RT_{NDT} shift is enhanced by certain chemical elements (such as copper and nickel).

Predicted ΔRT_{NDT} values are derived considering the effect of fluence and copper and nickel content for the reactor vessel steels exposed to 550°F temperature. U.S. NRC Regulatory Guide 1.99 is used in calculating adjusted reference temperature. Since the AP1000 cold leg temperature exceeds 525°F (minimum steady-state temperature is 535°F at 100% power, thermal design flow, and 10% tube plugging), the procedures of Regulatory Guide 1.99 for nominal embrittlement apply. The heatup and cooldown curves are developed considering a sufficient magnitude of radiation embrittlement so that no unirradiated ferritic materials in other components of the reactor coolant system will be limiting in the analysis.

The pressure-temperature curves are developed considering a radiation embrittlement of up to 54 effective full power years (EFPY) consistent with the plant design objective of 60 years with

90 percent availability. Copper, nickel contents and initial RT_{NDT} for materials in the reactor vessel beltline region and the reactor vessel flange and the closure head flange region are shown in Tables 5.3-1 and 5.3-3. The operating curves are developed with the methodology given in Reference 6, which is in accordance with 10 CFR 50, Appendix G with the following exceptions:

1.   The fluence values used are calculated fluence values (i.e., comply with Regulatory Guide 1.190), not the best-estimate fluence values.

2.   The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641 (which covers Code Cases N-640 and N-588).

3.   The 1996 Version of Appendix G to Section XI is used rather than the 1989 version.

The curves are applicable up to 54 effective full-power years. These curves, shown in Figures 5.3-2 and 5.3-3, are generic curves for the AP1000 reactor vessel design and they are limiting curves based on copper and nickel material composition (Reference 9). These curves are applicable as long as the following criteria are met:

- 10 CFR 50, Appendix G as related to pressure-temperature remains unchanged,

- Adjusted Reference Temperatures at 1/4T and 3/4T locations remain below the bases of Figures 5.3-2 and 5.3-3

The results of the material surveillance program described in subsection 5.3.2.6 will be used to verify the validity of ΔRT_{NDT} used in the calculation for the development of heatup and cooldown curves. The projected fluence, copper, and nickel contents along with the RT_{NDT} calculation will be adjusted if necessary, from time to time using the surveillance capsule results. This may require the development of new heatup and cooldown curves.

Higher rates of temperature changes when the reactor coolant system pressure is at or above the operating pressure do not impact the determination of the proper curve to use. Figure 5.3-2 also includes a curve for the leak test limit at steady-state temperature and curves for the criticality limit for nuclear heatup.

Temperature limits for core operation, inservice leak and hydrotests are calculated in accordance with the ASME Code, Section III, Appendix G.

5.3.4   Reactor Vessel Integrity

5.3.4.1   Design

The reactor vessel is the high pressure containment boundary used to support and enclose the reactor core. It provides flow direction with the reactor internals through the core and maintains a volume of coolant around the core. The vessel is cylindrical, with a transition ring, hemispherical bottom head, and removable flanged hemispherical upper head. The vessel is fabricated by welding together the lower head, the transition ring, the lower shell, and the upper shell. The upper shell contains the penetrations from the inlet and outlet nozzles and direct vessel injection

nozzles. The closure head is fabricated with a head dome and bolting flange. The upper head has penetrations for the control rod drive mechanisms, the incore instrumentation, head vent, and support lugs for the integrated head package.

The reactor vessel (including closure head) is approximately 40 feet long and has an inner diameter at the core region of 159 inches. The total weight of the vessel (including closure head and CRDMs) is approximately 417 tons. Surfaces which can become wetted during operation and refueling are clad to a nominal 0.22 inches of thickness with stainless steel welded overlay which includes the upper shell top surface but not the stud holes. The AP1000 reactor vessel's design objective is to withstand the design environment of 2500 psi and 650°F for 60 years. The major factor affecting vessel life is radiation degradation of the lower shell.

As a safety precaution, there are no penetrations below the top of the core. This eliminates the possibility of a loss of coolant accident by leakage from the reactor vessel which could allow the core to be uncovered. The core is positioned as low as possible in the vessel to limit reflood time in an accident. The main radial support system of the lower end of the reactor internals is accomplished by key and keyway joints to the vessel wall. At equally spaced points around the circumference, a clevis block is located on the reactor vessel inner diameter. A permanent cavity liner seal ring is attached to the top of the vessel shell for welding to the refueling cavity liner. To decrease outage time during refueling, access to the stud holes is provided to allow stud hole plugging with the head in place. By the use of a ring forging with an integral flange, the number of welds is minimized to decrease inservice inspection time.

The lower head has an approximate 6.5 feet inner spherical radius. The lower radial supports are located on the head at the elevation of the lower internals lower core support plate. The transition ring is welded to the lower shell course with the weld located outside the high fluence active core region. The lower shell is a ring forging about 8 inches thick with an inner diameter of 159 inches. The length of the shell is greater than 168 inches to place the upper shell weld outside of the active fuel region. The upper shell is a large ring forging. Included in this forging are four 22-inch inner diameter inlet nozzles, two 31-inch inner diameter outlet nozzles and two 6.81-inch inner diameter direct vessel injection nozzles (8-inch schedule 160 pipe connections). These nozzles are forged into the ring or are fabricated by "set in" construction. The inlet and outlet nozzles are offset axially in different planes by 17.5 inches. The injection nozzles are 100 inches down from the main flange and the outlet nozzles are 80 inches down and the inlet nozzles are 62.5 inches below the mating surface.

The closure head has a 77.5-inch inner spherical radius and a 188.0-inch O.D. outer flange. Cladding is extended across the bottom of the flange for refueling purposes. Forty-five, seven-inch diameter studs attach the head to the lower vessel and two metal o-rings are used for sealing. The upper head has sixty-nine 4-inch outer diameter penetrations for the control rod drive mechanism housings and eight penetrations for the incore instrumentation tubes.

The eight penetrations for the incore instrumentation tubes are Quickloc instrument nozzles, which are welded to the reactor vessel head. Up to six instrument thimble assemblies pass through each Quickloc instrument nozzle. The reactor vessel head penetration diameter is approximately 4.75 inches to the cladding. The material of the pressure boundary parts of the Quickloc are SA-182 – Type F304, SA-479 – Type 304 and Type 316, and UNS S21800. The Quickloc

instrument nozzle is welded to the Ni-Cr-Fe buttering on the weld buildup on the reactor vessel closure head. The Quickloc provides two pressure boundaries: 1) between the Quickloc plug and the incore instrument thimble assemblies (this pressure boundary is disassembled only when the instrument thimble is replaced) and 2) between the Quickloc plug and the instrument nozzle (this pressure boundary is disassembled when the reactor vessel closure head is removed). The Quickloc instrument nozzle pressure boundary parts are designed and fabricated to the requirements of the ASME Code, Section III Division 1, Subsection NB. The Quickloc internal, non-pressure boundary parts are designed to Subsection NG. The Quickloc assembly is a proven design, which was first installed in an operating plant in 1995. Since then, it has been installed on four additional operating plants.

The vessel is manufactured from low alloy steel plates and forgings to minimize size. The chemical content of the core region base material is specifically controlled. A surveillance program is used to monitor the radiation damage to the vessel material.

The four vessel supports are located beneath the inlet nozzles and the internals support ledge is machined into the top of the upper shell. The top of the upper shell contains the stud holes and has the sealing surface for the closure head. Inner and outer monitor tubes are provided through the shell to collect any leakage past the closure region o-rings.

The reactor vessel is designed and fabricated in accordance with the quality standards set forth in 10 CFR 50, General Design Criteria 1, 14, 30, and 31, and 50.55a; and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section III. Principal design parameters of the reactor vessel are given in Table 5.3-5. The vessel design and construction enables inspection in accordance with the ASME Code, Section XI.

Cyclic loads are introduced by normal power changes, reactor trips, and startup and shutdown operations. These design base cycles are selected for fatigue evaluation and constitute a conservative design envelope for the design life. Thermal stratification during passive core cooling system operation and natural circulation cooldown is considered by performing a thermal/flow analysis using computational fluid dynamics techniques. This analysis includes thermally-induced fluid buoyancy, heat transfer between the coolant and the metal of the vessel and internals and uses thermal/flow boundary conditions based on an existing thermal/hydraulic transient analysis of the primary reactor coolant system. This analysis provides temperature maps that are used to evaluate thermal stresses.

Analysis proves that the vessel is in compliance with the fatigue and stress limits of the ASME Code, Section III. The loadings and transients specified for the analysis are based on the most severe conditions expected during service. The heatup and cooldown rates imposed by plant operating limits are 100°F per hour for normal operations.

5.3.4.2   Materials of Construction

The materials used in the fabrication of the reactor vessel are discussed in subsection 5.2.3.

5.3.4.3   Fabrication Methods

The fabrication methods used in the construction of the reactor vessel are discussed in subsection 5.3.2.2.

5.3.4.4   Inspection Requirements

The nondestructive examinations performed on the reactor vessel are described in subsection 5.3.2.3.

5.3.4.5   Shipment and Installation

The reactor vessel is shipped in a horizontal position on a shipping skid with a vessel-lifting truss assembly. All vessel openings are sealed to prevent the entrance of moisture, and an adequate quantity of desiccant bags is placed inside the vessel. These are usually placed in a wire mesh basket attached to the vessel cover. All carbon steel surfaces are protected with a temporary protective covering before shipment.

The closure head is also shipped with a shipping cover and skid. An enclosure attached to the ventilation shroud support ring protects the control rod mechanism housings. All head openings are sealed to prevent the entrance of moisture, and an adequate quantity of desiccant bags is placed inside the head. These are placed in a wire mesh basket attached to the head cover. All carbon steel surfaces are protected with a temporary protective covering before shipment.

5.3.4.6   Operating Conditions

Operating limitations for the reactor vessel are presented in subsection 5.3.3 and in the technical specifications.

In addition to the analysis of primary components discussed in subsection 3.9.1.4, the reactor vessel is further qualified to ensure against unstable crack growth under faulted conditions. Safeguard actuation following a loss-of-coolant, tube rupture or other similar emergency or faulted event produces relatively high thermal stresses in regions of the reactor vessel which come into contact with water from the passive core cooling system. Primary consideration is given to these areas, including the reactor vessel beltline region and the reactor vessel primary coolant nozzles, to ensure the integrity of the reactor vessel under these severe postulated transients. TMI Action Item II.K.2.13, is satisfied upon submittal of RT_{NDT} values which are below the pressurized thermal shock (PTS) rule screening values. The results given in Table 5.3-3 show that the issue is resolved.

For the beltline region, the NRC staff concluded that conservatively calculated screening criterion values of RT_{NDT} less than 270°F for plate material and axial welds, and less than 300°F for circumferential welds, present an acceptably low risk of vessel failure from pressurized thermal shock events. These values were chosen as the screening criterion in the pressurized thermal shock rule for 10 CFR 50.34 (new plants) and 10 CFR 50.61 (operating plants). The conservative methods chosen by the NRC staff for the calculation of RT_{PTS} for the purpose of comparison with the screening criterion is presented in paragraph (b)(2) of 10 CFR 50.61. Details of the analysis

method and the basis for the pressurized thermal shock rule can be found in SECY-82-465 (Reference 4).

The revised pressurized thermal shock rule, (10 CFR 50.61), effective June 14, 1991 makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99.

The reactor vessel beltline materials are specified in subsection 5.3.2. Evaluation of the AP1000 reactor vessel material showed that even at the fluence level which results in the highest RT_{PTS} value, this value is well below the screening criteria of 270°F. RT_{PTS} is RT_{NDT}, the reference nil ductility transition temperature as calculated by the method chosen by the NRC staff as presented in paragraph (b)(2) of 10 CFR 50.61, and the pressurized thermal shock rule. The pressurized thermal shock rule states that this method of calculating RT_{PTS} should be used in reporting values used to compare pressurized thermal shock to the above screening criterion set in the pressurized thermal shock rule. The screening criteria will not be exceeded using the method of calculation prescribed by the pressurized thermal shock rule for the vessel design objective. The material properties, and initial RT_{NDT} and end-of-life RT_{PTS} requirements and predictions are in Tables 5.3-1 and 5.3-3. The materials that are exposed to high fluence levels at the beltline region of the reactor vessel are subject to the pressurized thermal shock rule. These materials are a subset of the reactor vessel materials identified in subsection 5.3.2.

The principles and procedures of linear elastic fracture mechanics (LEFM) are used to evaluate thermal effects in the regions of interest. The linear elastic fracture mechanics approach to the design against failure is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption employed in linear elastic fracture mechanics is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack tip to the applied stress on the structure, the material properties, and the size of defect necessary to cause failure.

5.3.4.7   Inservice Surveillance

The internal surfaces of the reactor vessel are accessible for periodic inspection. Visual and/or nondestructive techniques are used. During refueling, the vessel cladding is capable of being inspected in certain areas of the upper shell above the primary coolant inlet nozzles, and if deemed necessary, the core barrel is capable of being removed, making the entire inside vessel surface accessible.

The closure head is examined visually during each refueling. Optical devices permit a selective inspection of the cladding, control rod drive mechanism nozzles, and the gasket seating surface. Access to the top head surface is provided by 7 ports around the circumference of the integrated head package shroud and by multiple removable insulation panels, which interface with the head under the integrated head package shroud. Both the ports and the insulation panels provide access to the bare vessel head, and CRDM and instrumentation penetrations for use of a remote, mobile visual inspection manipulator to perform a 360° inspection around each penetration. The head insulation is a stand-off design with a minimum offset from the head surface of 3 inches.

The knuckle transition piece, which is the area of highest stress of the closure head, is accessible on the outer surface for visual inspection, dye penetrant or magnetic particle testing, and ultrasonic testing. The closure studs and nuts can be inspected periodically using visual, magnetic particle, and ultrasonic techniques.

The closure studs, nuts, washers, and the vessel flange seal surface, as well as the full-penetration welds in the following areas of the installed reactor vessel, are available for nondestructive examination:

- Vessel shell, from the inside surface.

- Primary coolant nozzles, from the inside surface. Only partial outside diameter coverage is provided.

- Closure head, from the inside surface; bottom head, from the inside surface.

- Field welds between the reactor vessel nozzle safe ends and the main coolant piping, from the inside surface.

The design considerations which have been incorporated into the system design to permit the above inspection are as follows:

- Reactor internals are completely removable. The tools and storage space required to permit removal of the reactor internals are provided.

- The closure head is stored on a stand on the reactor operating deck during refueling to facilitate direct visual inspection.

- Reactor vessel studs, nuts, and washers can be removed to dry storage during refueling.

- Access is provided to the reactor vessel nozzle safe ends. The insulation covering the nozzle-to-pipe welds may be removed.

Because radiation levels and remote underwater accessibility limits access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for the periodic nondestructive tests which are required by the ASME Code inservice inspection requirements. These are as follows:

- Shop ultrasonic examinations are performed on internally clad surfaces to an acceptance and repair standard to provide an adequate cladding bond to allow later ultrasonic testing of the base metal from the inside surface. The size of cladding bond defect allowed is 0.25 inch by 0.75 inch with the greater direction parallel to the weld in the region bounded by 2T (T = wall thickness) on both sides of each full-penetration pressure boundary weld. Unbounded areas exceeding 0.442 square inches (0.75-inch diameter) in other regions are rejected.

- The design of the reactor vessel shell is an uncluttered cylindrical surface to permit future positioning of the test equipment without obstruction.

- The weld-deposited clad surface on both sides of the welds to be inspected is specifically prepared to ensure meaningful ultrasonic examinations.

- During fabrication, full-penetration ferritic pressure boundary welds are ultrasonically examined in addition to code examinations.

- After the shop hydrostatic testing, full-penetration ferritic pressure boundary welds (with the exception of the closure head welds), as well as the nozzles to safe end welds, are ultrasonically examined from both the inside and outside diameters in addition to ASME Code, Section III requirements.

- Preservice examinations for the closure head will include a baseline top-of-the head visual examination; ultrasonic examinations of the inside diameter surface of each vessel head penetration; eddy current examinations of the surface of head penetration welds, the outside diameter surface of the vessel penetrations, and the inside diameter surface of the penetrations; and post-hydro liquid penetrant examinations of accessible surfaces that have undergone preservice inspection eddy current examinations.

The vessel design and construction enables inspection in accordance with the ASME Code, Section XI.

5.3.5   Reactor Vessel Insulation

5.3.5.1   Reactor Vessel Insulation Design Bases

Reactor vessel insulation is provided to minimize heat losses from the primary system. Nonsafety-related reflective insulation similar to that in use in current pressurized water reactors is utilized. The AP1000 reactor vessel insulation contains design features to promote in-vessel retention following severe accidents. In the unlikely event of a beyond design basis accident, the reactor cavity is flooded with water, and the reactor vessel insulation allows heat removal from core debris via boiling on the outside surface of the reactor vessel. The reactor vessel insulation permits a water layer next to the reactor vessel to promote heat transfer from the reactor vessel. This is accomplished by providing:

- A means of allowing water free access to the region between the reactor vessel and insulation.

- A means to allow steam generated by water contact with the reactor vessel to escape from the region surrounding the reactor vessel.

- The insulation support frame and the insulation panels form a structurally reliable flowpath for the water and steam.

The reactor vessel insulation and its supports are designed to withstand bounding pressure differentials across the reactor vessel insulation panels during the period that the reactor vessel is externally flooded with water and the core heat is removed from the vessel wall by water and generated steam is vented. This is accomplished by providing steam vents with a minimum flow area of 12 ft^2 from the vessel insulation annular space. The flow path from the reactor loop compartment to the reactor cavity provides an open flow path for water to flood the reactor cavity. The reactor vessel insulation water inlets are designed to minimize the pressure drop during ex-vessel cooling to permit water inflow to cool the vessel.

5.3.5.2   Description of Insulation

A schematic of the reactor vessel, the vessel insulation and the reactor cavity is shown in Figure 5.3-7. The insulation is mounted on a structural frame that is supported from the wall and floor of the reactor cavity. The insulation panels are designed to have a minimum gap between the insulation and reactor vessel not less than 2 inches when subjected to the dynamic loads in the direction towards the vessel that result during ex-vessel cooling.

The bottom portion of the vessel insulation is constructed to provide a flow channel conducive for heat removal.

The structural frame supporting the insulation is designed to withstand the bounding severe accident loads while maintaining the flow path. The fasteners holding the insulation panels to the frame are also designed for these loads.

At the bottom of the insulation are water inlet assemblies. Each water inlet assembly is normally closed to prevent an air circulation path through the vessel insulation. The inlet assemblies are self-actuating passive devices. The inlet assemblies open when the cavity is filled with water. This permits ingress of water during a severe accident, while preventing excessive heat loss during normal operation.

The total flow area of the water inlet assemblies has sufficient margin to preclude significant pressure drop during ex-vessel cooling during a severe accident. The minimum total flow area for the water inlets assemblies is 6 ft^2. Due to the relatively low approach velocities in the flow paths leading to the reactor cavity, the grating over the vertical access tunnel, the design of the doorway between the reactor coolant drain tank compartment and the reactor cavity, the low flow velocities approaching the water inlet assemblies, and the relatively large minimum flow area through each water inlet assembly, at least 7 in^2, the water inlet assemblies and the steam flow path are not susceptible to clogging from debris inside containment.

Multiple steam vents in the nozzle gallery provide a flow path for the steam/water within the reactor vessel/insulation annular space to flow back to the containment flood-up region. The steam vents provide 12 ft^2 minimum flow area for steam/water to exit the annular space. Each of the steam vents has a door that will be opened by the steam/water flow generated under the insulation with the cavity filled with water, but which remains in place when only normal air cooling flow is operating.

Extensive maintenance of the vessel insulation is not normally required. Periodic verification of the vessel insulation moving parts can be performed during refueling outages.

5.3.5.3   Description of External Vessel Cooling Flooded Compartments

Ex-vessel cooling during a severe accident is provided by flooding the reactor coolant system loop compartment including the vertical access tunnel, the reactor coolant drain tank room, and the reactor cavity. Water from these compartments replenishes the water that comes in contact with the reactor vessel and is boiled and vented to containment. The opening between the vertical access tunnel and the reactor coolant drain tank room is approximately 100 ft^2. Removable steel grating is provided over the inlet to the vertical access tunnel to restrict access to the lower compartments. This grating precludes large debris from being transported into the reactor cavity during ex-vessel cooling scenarios. Figure 5.3-8 depicts the flooded compartments that provide the water for ex-vessel cooling. The doorway between the reactor cavity compartment and the reactor coolant drain tank room consists of a normally closed door and a damper above the door. The door and damper arrangement, shown in Figure 5.3-9, maintains the proper air flow through the reactor cavity during normal operation. The damper prevents air from flowing into the reactor coolant drain tank compartment, but opens to permit flooding of the reactor cavity from the reactor coolant drain tank compartment. The damper opening has a minimum flow area of 8 ft^2 and is not susceptible to clogging from debris that can pass through the grating over the inlet to the vertical access tunnel. It is constructed of light-weight material to minimize the force necessary to open the damper and permit flooding and continued water flow through the opening during ex-vessel cooling. The damper provides an acceptable pressure drop through the opening during ex-vessel cooling.

DCD subsection 6.3.2.1.3 discusses post-accident operation of the passive core cooling system, which operates to flood the reactor cavity following an accident. DCD subsection 9.1.3 discusses the connections from the refueling cavity to the steam generator compartment that facilitate flooding of the reactor cavity following an accident.

5.3.5.4   Determination of Forces on Insulation and Support System

The forces that may be expected in the reactor cavity region of the AP1000 plant during a core damage accident in which the core has relocated to the lower head and the reactor cavity is reflooded can be based on test results from the ULPU test program (Reference 5). The particular configuration (Configuration V) reviewed closely models the full-scale AP1000 geometry of water in the region near the reactor vessel, between the reactor vessel and the reactor vessel insulation. The ULPU tests provide data on the pressure generated in the region between the reactor vessel and reactor vessel insulation. These data, along with observations and conclusions from heat transfer studies, are used to develop the functional requirements with respect to in-vessel retention for the reactor vessel insulation and support system. Interpretation of data collected from ULPU Configuration V experiments in conjunction with the static head of water that would be present in the AP1000 is used to estimate forces acting on the rigid sections of insulation. The ULPU V test results indicate that the pressure variations in the flow channel between the vessel and the insulation are on the order of plus/minus 0.5 meters of water. Fast Fourier Transform analysis of the ULPU V pressure data is also included in the ULPU V test report. This analysis shows that the dominant frequency of the pressure variations is less than about 2 Hz. The natural frequency of the insulation structure is expected to be well above 2 Hz.

5.3.5.5   Design Evaluation

A structural analysis of the AP1000 reactor cavity insulation system was performed that demonstrates that it meets the functional requirements discussed above. The analysis encompasses the insulation and support system and includes a determination of the stresses in support members, bolts, insulation panels and welds, as well as deflection of support members and insulation panels.

Loads on the insulation and the support structure include hydrostatic loads and dynamic loads from boiling. These loads are of the same order as those analyzed for AP600, and the results of the AP1000 analysis show that the insulation is able to meet its functional requirements. The reactor vessel insulation provides an engineered pathway for water-cooling the vessel and for venting steam from the reactor cavity. These results were also compared to the available test data.

The reactor vessel insulation is purchased equipment. The purchase specification for the reactor vessel insulation design required confirmatory static load analyses.

5.3.6   Combined License Information

5.3.6.1   Pressure-Temperature Limit Curves

The pressure-temperature curves shown in Figures 5.3-2 and 5.3-3 are generic curves for AP1000 reactor vessel design, and they are the limiting curves based on copper and nickel material composition. However, for a specific AP1000, these curves will be plotted based on material composition of copper and nickel. Use of plant-specific curves will be addressed by the Combined License holder during procurement and fabrication of the reactor vessel. As noted in the bases to Technical Specification 3.4.14, use of plant-specific curves requires evaluation of the LTOP system. This includes an evaluation of the setpoint pressure for the RNS relief valve by the Combined License holder to determine if the setpoint pressure needs to be changed based on the plant-specific pressure-temperature curves. The development of the plant-specific curves and evaluation of the setpoint pressure are required prior to fuel load.

5.3.6.2   Reactor Vessel Materials Surveillance Program

The Combined License applicant will address a reactor vessel reactor material surveillance program based on subsection 5.3.2.6.

5.3.6.3   Surveillance Capsule Lead Factor and Azimuthal Location Confirmation

The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-023 (Reference 7), and the applicable changes are incorporated into the DCD. No additional work is required by the Combined License applicant to address the Combined License information requested in this subsection.

The following words represent the original Combined License Information Item commitment, which has been addressed as discussed above:

The Combined License applicant will address confirmation of the surveillance capsule lead factors and azimuthal locations through an analysis which includes modeling of the capsule/holder.

5.3.6.4   Reactor Vessel Materials Properties Verification

5.3.6.4.1   The Combined License holder will complete prior to fuel load verification of plant-specific belt line material properties consistent with the requirements in subsection 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification will include a pressurized thermal shock evaluation based on as-procured reactor vessel material data and the projected neutron fluencies for the plant design objective of 60 years. This evaluation report will be submitted for NRC staff review.

5.3.6.4.2   The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-060 (Reference 8), and the applicable changes are incorporated into the DCD. No additional work is required by the Combined License applicant to address the Combined License information requested in this subsection.

The following words represent the original Combined License Information Item commitment, which has been addressed as discussed above:

The verification will include structural analysis of the AP1000 reactor vessel insulation and support structure.

5.3.6.5   Reactor Vessel Insulation

The Combined License information requested in this subsection has been fully addressed in Reference 8 (APP-GW-GLR-060). No additional work is required by the Combined License applicant to address the Combined License information requested in this subsection.

The following words represent the original Combined License Information Item commitment, which has been addressed as discussed above:

The Combined License applicant will address verification that the reactor vessel insulation is consistent with the design bases established for in-vessel retention. The ULPU Configuration V test data is suitable to be used to develop the design loads for the AP1000 reactor vessel insulation design.

5.3.6.6   Inservice Inspection of Reactor Vessel Head Weld Buildup

The Combined License holder will establish an inservice inspection program prior to fuel load. The inservice inspection program will include the performance of a 100-percent volumetric examination of the weld buildup on the reactor vessel head for the instrumentation penetrations (Quickloc) conducted once during each 120-month inspection interval in accordance with the ASME Code, Section XI. The weld buildup shall meet the acceptance standards of ASME Code, Section XI, IWB-3514. Personnel performing examinations and the ultrasonic examination systems shall be qualified in accordance with ASME Code, Section XI, Appendix VIII. Alternatively, the Combined License holder may develop an alternative inspection in conjunction

with the voluntary consensus standards bodies (that is, ASME) and submit to the NRC for approval.

5.3.7   **References**

1.   ASTM E-185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

2.   Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," United States Nuclear Regulatory Commission, Office of Nuclear Reactor Research, March, 2001.

3.   WCAP-15557, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," S. L. Anderson, August 2000.

4.   NRC Policy Issue, "Pressurized Thermal Shock," SECY-82-465, November 23, 1982.

5.   Theofanous, T.G., et al., "Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility," CRSS-03/06, June 2003.

6.   WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.

7.   APP-GW-GLR-023, "Surveillance Capsule Lead Factor and Azimuthal Location Confirmation," Westinghouse Electric Company LLC.

8.   APP-GW-GLR-060, "Reactor Vessel Insulation System – Verification of In-Vessel Retention Design Bases," Westinghouse Electric Company LLC.

9.   APP-RXS-Z0R-001, Revision 2, "AP1000 Generic Pressure Temperature Limits Report," F. C. Gift, September 2008.

Table 5.3-1		
MAXIMUM LIMITS FOR ELEMENTS OF THE REACTOR VESSEL		
Element	**Beltline Forging (percent)**	**As Deposited Weld Metal (percent)**
Copper	0.06	0.06
Phosphorus	0.01	0.01
Vanadium	0.05	0.05
Sulfur	0.01	0.01
Nickel	0.85	0.85

	Table 5.3-2 **REACTOR VESSEL QUALITY ASSURANCE PROGRAM**			
	RT[a]	**UT**[a]	**PT**[a]	**MT**[a]
Forgings				
Flanges		Yes		Yes
Studs and nuts		Yes		Yes
CRDM head adapter tube		Yes	Yes	
Instrumentation tube		Yes	Yes	
Main nozzles		Yes		Yes
Nozzle safe ends		Yes	Yes	
Shell sections		Yes		Yes
Heads		Yes		Yes
Plates		Yes		Yes
Weldments				
Head and shell	Yes	Yes		Yes
CRDM head adapter to closure head connection			Yes	
Instrumentation tube to closure head connection			Yes	
Main nozzle	Yes	Yes		Yes
Cladding		Yes	Yes	
Nozzle to safe ends	Yes	Yes	Yes	
CRDM head adapter flange to CRDM head adapter tube	Yes		Yes	
All full-penetration ferritic pressure boundary welds accessible after hydrotest		Yes		Yes
Full-penetration nonferritic pressure boundary welds accessible after hydrotest a. Nozzle to safe ends		Yes	Yes	
Seal ledge				Yes
Head lift lugs				Yes
Core pad welds			Yes	
Flow skirt support lugs weld buildup		Yes	Yes	

Notes:

a. RT - Radiographic
    UT - Ultrasonic
    PT - Dye penetrant
    MT - Magnetic particle

Base metal weld repairs as a result of UT, MT, RT, and/or PT indications are cleared by the same nondestructive examination technique/procedure by which the indications were found. The repairs meet applicable Section III requirements.

In addition, UT examination in accordance with the in process/posthydro UT requirements is performed on base metal repairs in the core region and base metal repairs in the inservice inspection zone (1/2 T).

Table 5.3-3 END-OF-LIFE RT_{NDT} AND UPPER SHELF ENERGY PROJECTIONS				
	Unirradiated		**End-of-life (54 EFPY)**	
	RT_{NDT} **(°F)**	**USE (ft-lb)**	**USE (ft-lb) 1/4T**	RT_{PTS} **(°F)**
Beltline Forging	-10	> 75	> 50	< 270[2]
Head	10	N/A	N/A	N/A
Flange	10	N/A	N/A	N/A
Weld	10	N/A	N/A	N/A
Beltline Weld	-20	> 75	> 50	< 300[2]

Notes:
1. The minimum unirradiated upper shelf energy for beltline base metal is for the transverse direction.
2. End-of-Life RT_{PTS} requirements shown. End-of-Life RT_{PTS} (also equals RT_{NDT}) will be determined for as-built material. The preliminary RT_{PTS} for the AP1000 reactor vessel beltline forging and beltline weld are 94°F and 148°F, respectively.

Table 5.3-4			
REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM			
Capsules S, T, U, V, W, X, Y, and Z			
Material	**Charpy Specimens**	**Tensile Specimens**	**1/2T-CT Specimens**
Forging (tangential)	15	3	2
Forging (axial)	15	3	2
Weld Metal	15	3	2
Heat Affected Zone (HAZ)	15	–	–

Table 5.3-5	
REACTOR VESSEL DESIGN PARAMETERS	
(approximate values)	
Design pressure (psig)	2485
Design temperature (°F)	650
Overall height of vessel and closure head, bottom head outside diameter to top of control rod mechanism (ft-in.)	45-9
Number of reactor closure head studs	45
Diameter of reactor closure head/studs, (in.)	7
Outside diameter of closure head flange (in.)	188
Inside diameter of flange (in.)	148.81
Outside diameter at shell (in.)	176
Inside diameter at shell (in.)	159
Inlet nozzle inside diameter (in.)	22
Outlet nozzle inside diameter (in.)	31
Clad thickness, nominal (in.)	0.22
Lower head thickness, minimum (in.)	6
Vessel beltline thickness, minimum (in.)	8
Closure head thickness (in.)	6.25
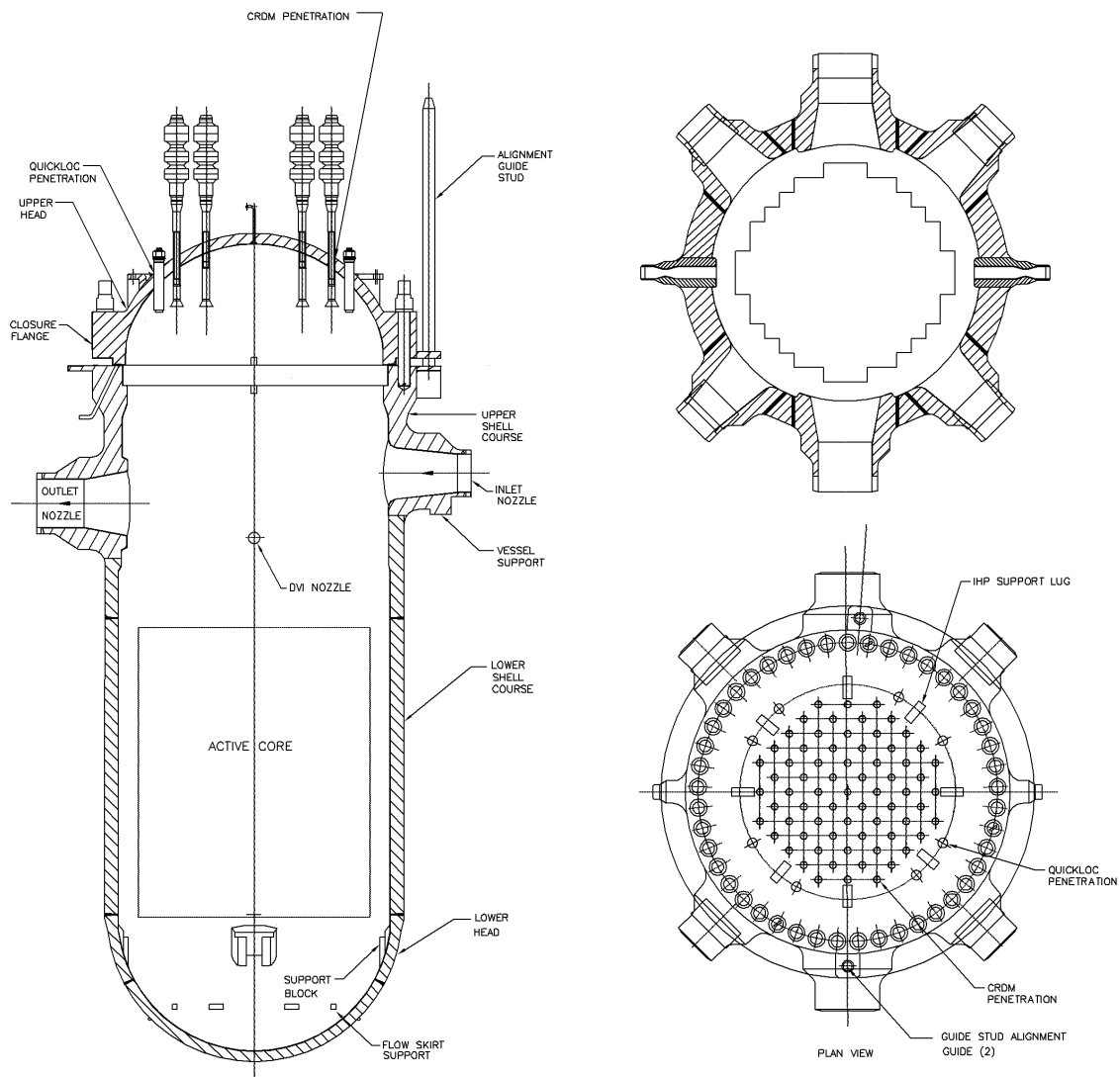
Figure 5.3-1

Reactor Vessel

AP1000: 54 EFPY Curve, using 1996 App. G w/Kic, w/ flange, w/o margins; dated August 24, 2006
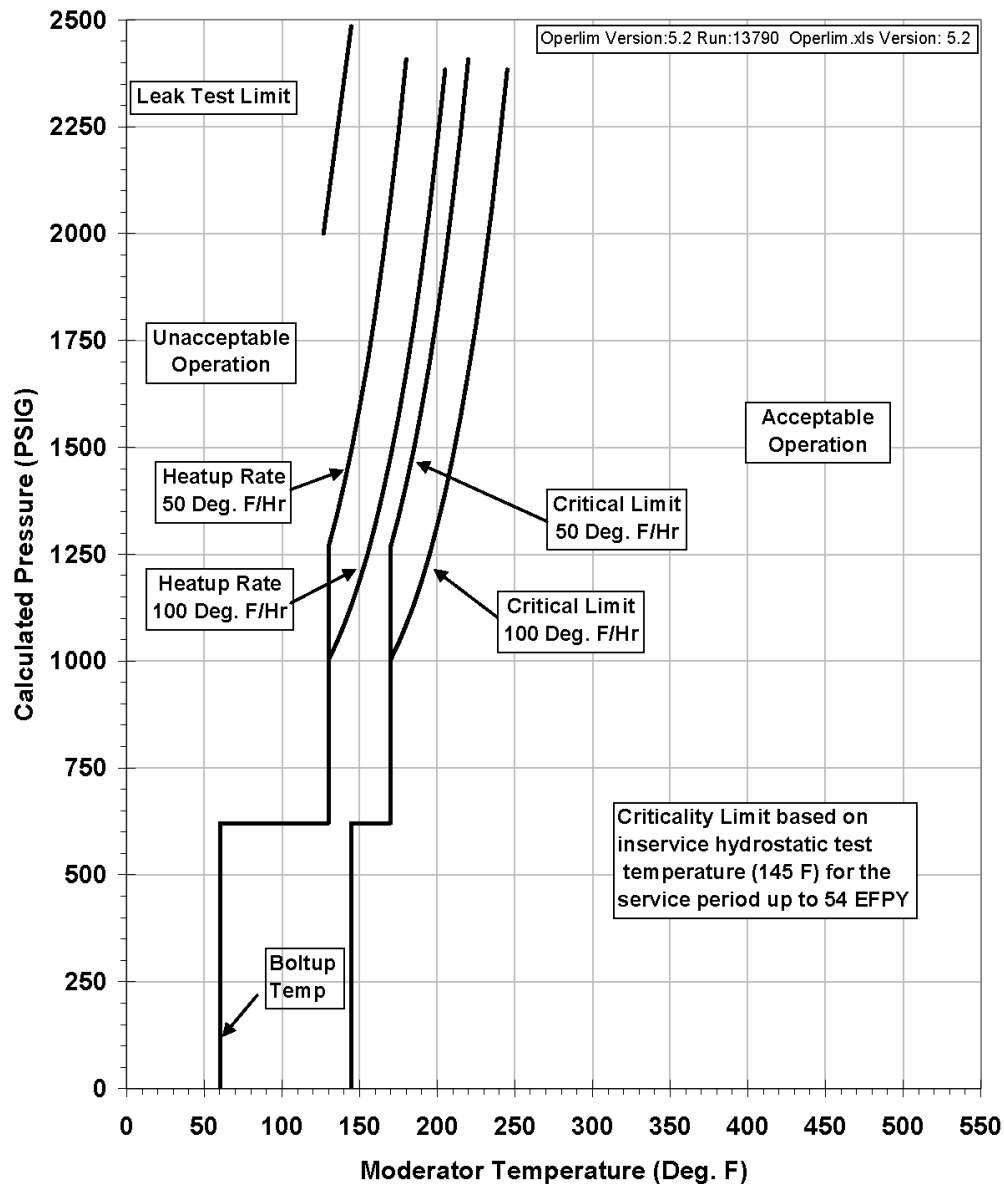
Figure 5.3-2

AP1000 Reactor Coolant System Heatup Limitations (Heatup Rate Up to 50° and 100°F/hour) Representative for the First 54 EFPY (Without Margins for Instrumentation Errors)
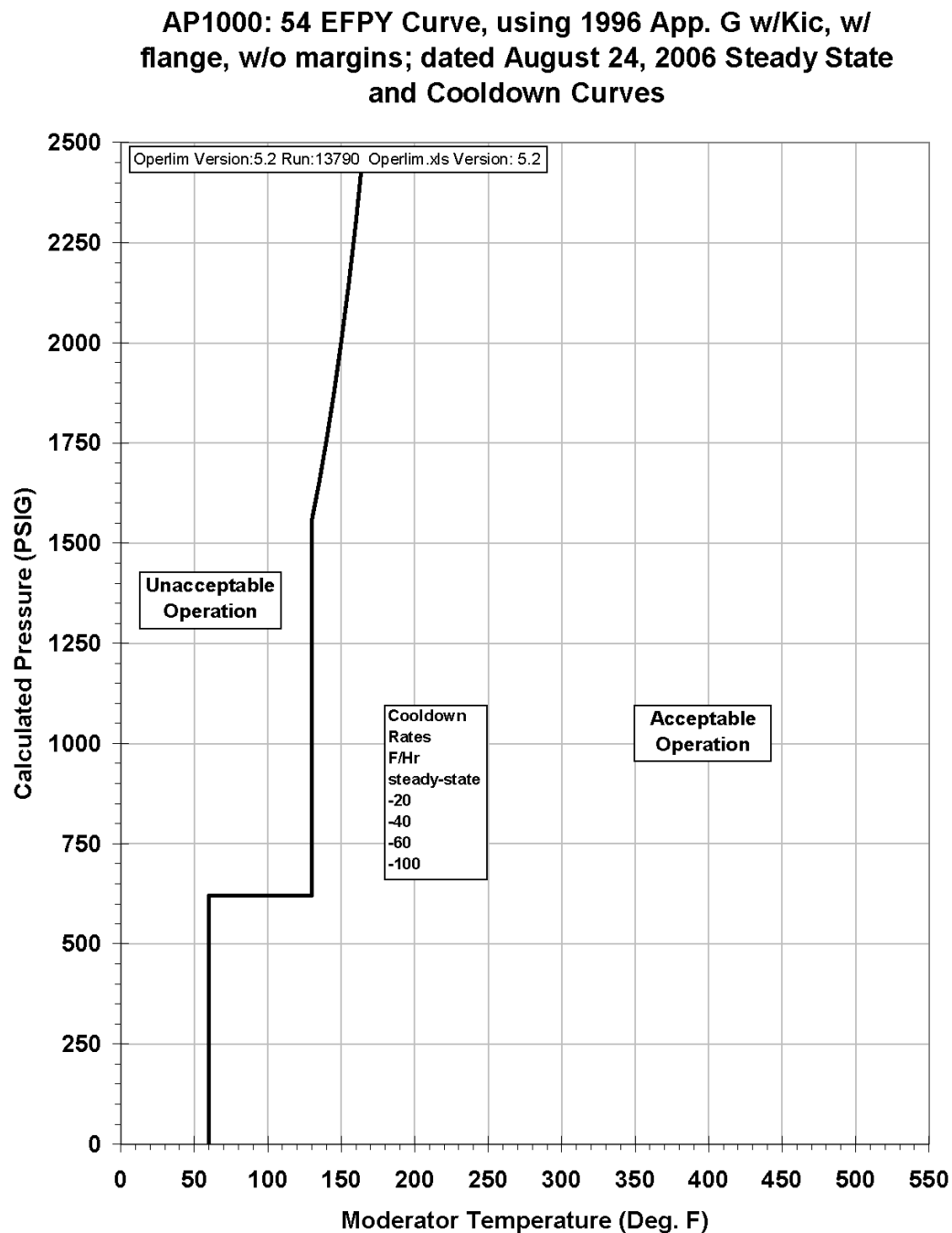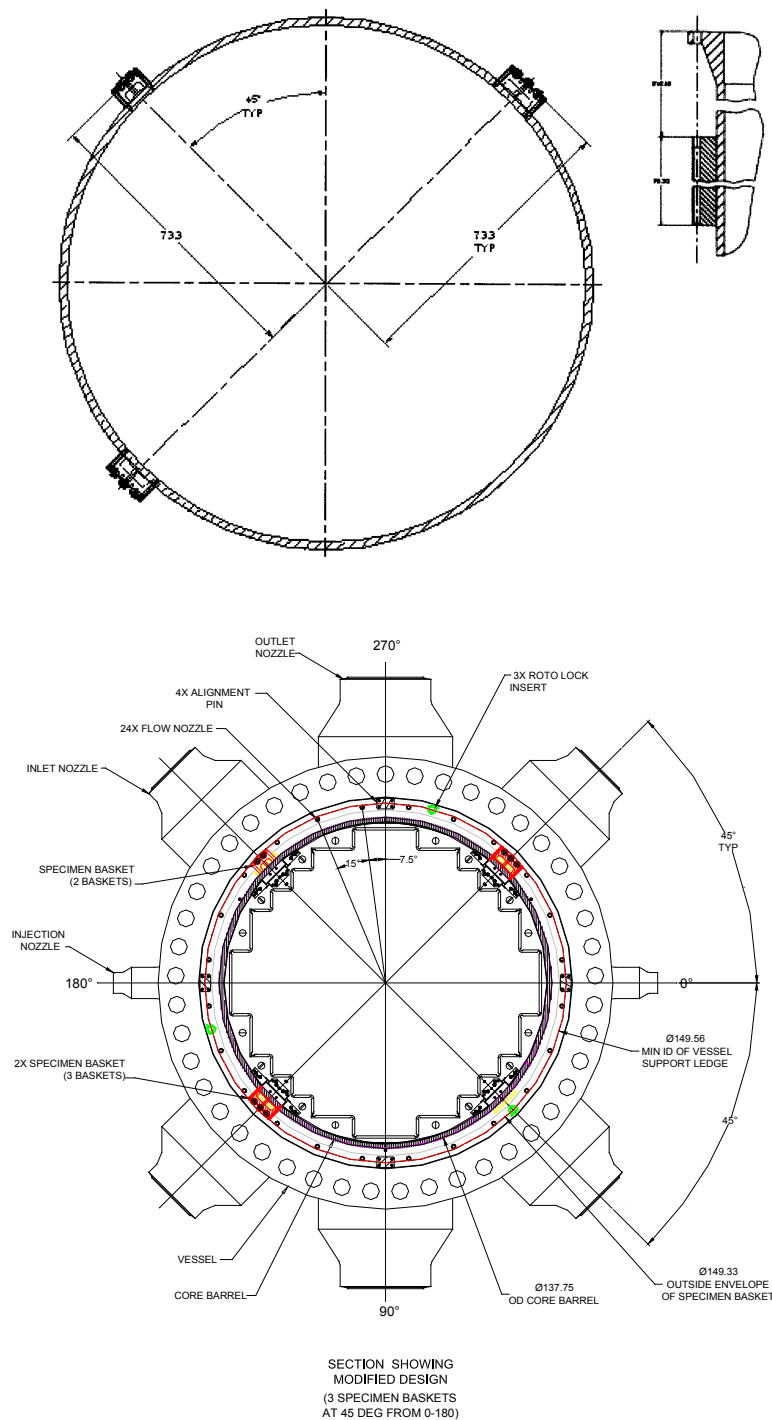
AP1000: 54 EFPY Curve, using 1996 App. G w/Kic, w/ flange, w/o margins; dated August 24, 2006 Steady State and Cooldown Curves

Figure 5.3-3

AP1000 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 50° and 100°F/hour) Representative for the First 54 EFPY (Without Margins for Instrumentation Errors)

Figure 5.3-4

AP1000 Reactor Vessel Surveillance Capsules Locations
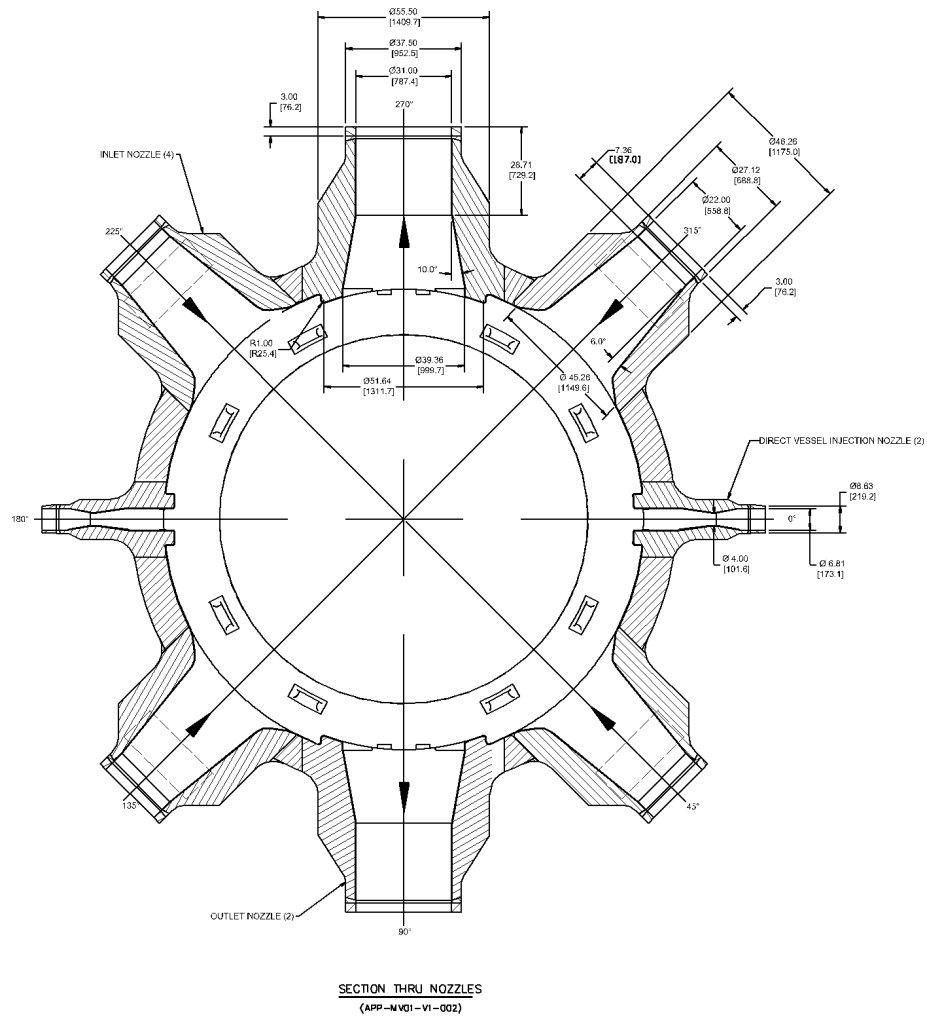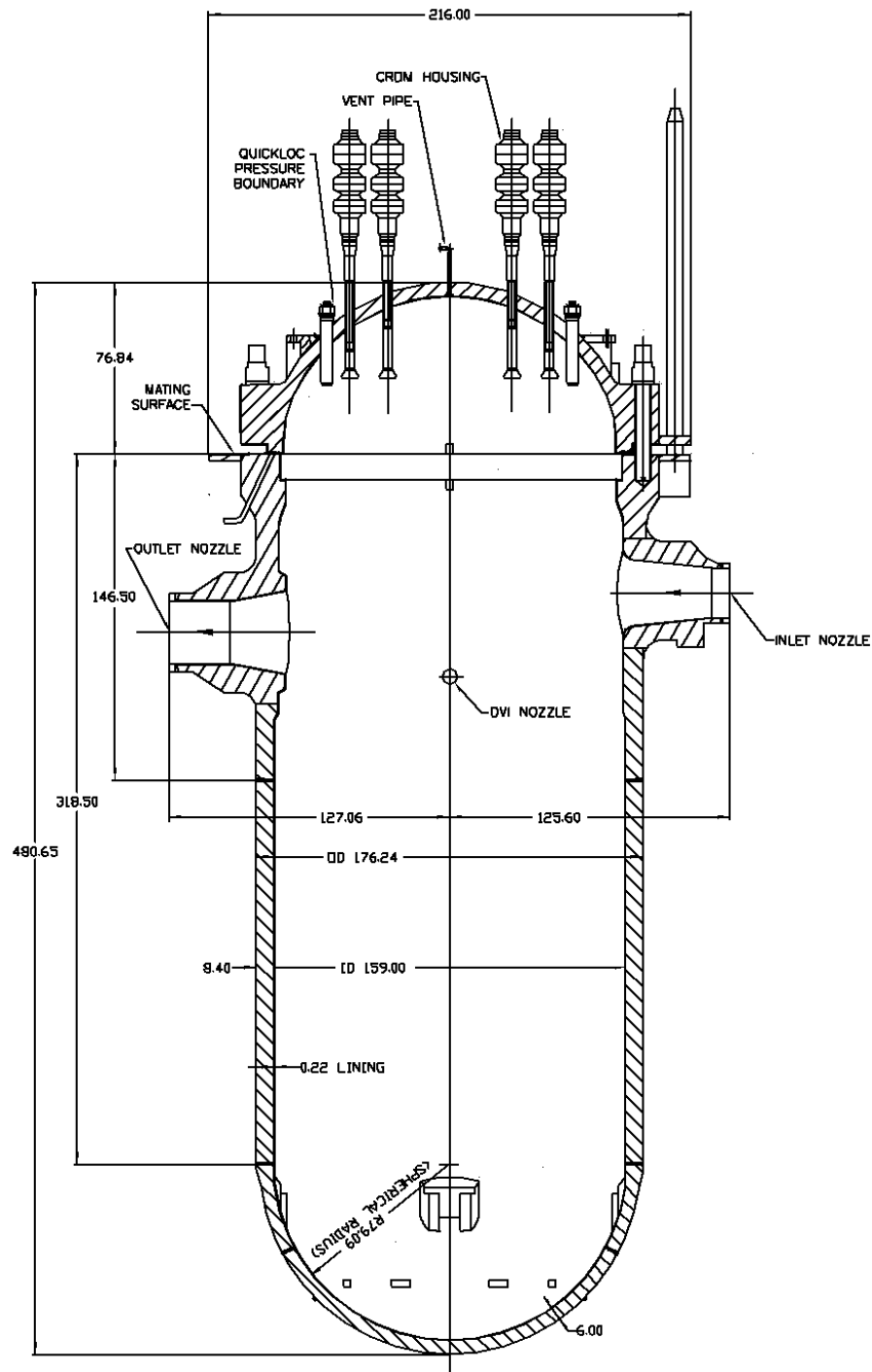
Figure 5.3-5

Reactor Vessel Key Dimensions
Plan View

Figure 5.3-6

**Reactor Vessel Key Dimensions,
Side View**

STEAM VENT

REACTOR VESSEL

STEAM VENT

CORE

REACTOR VESSEL
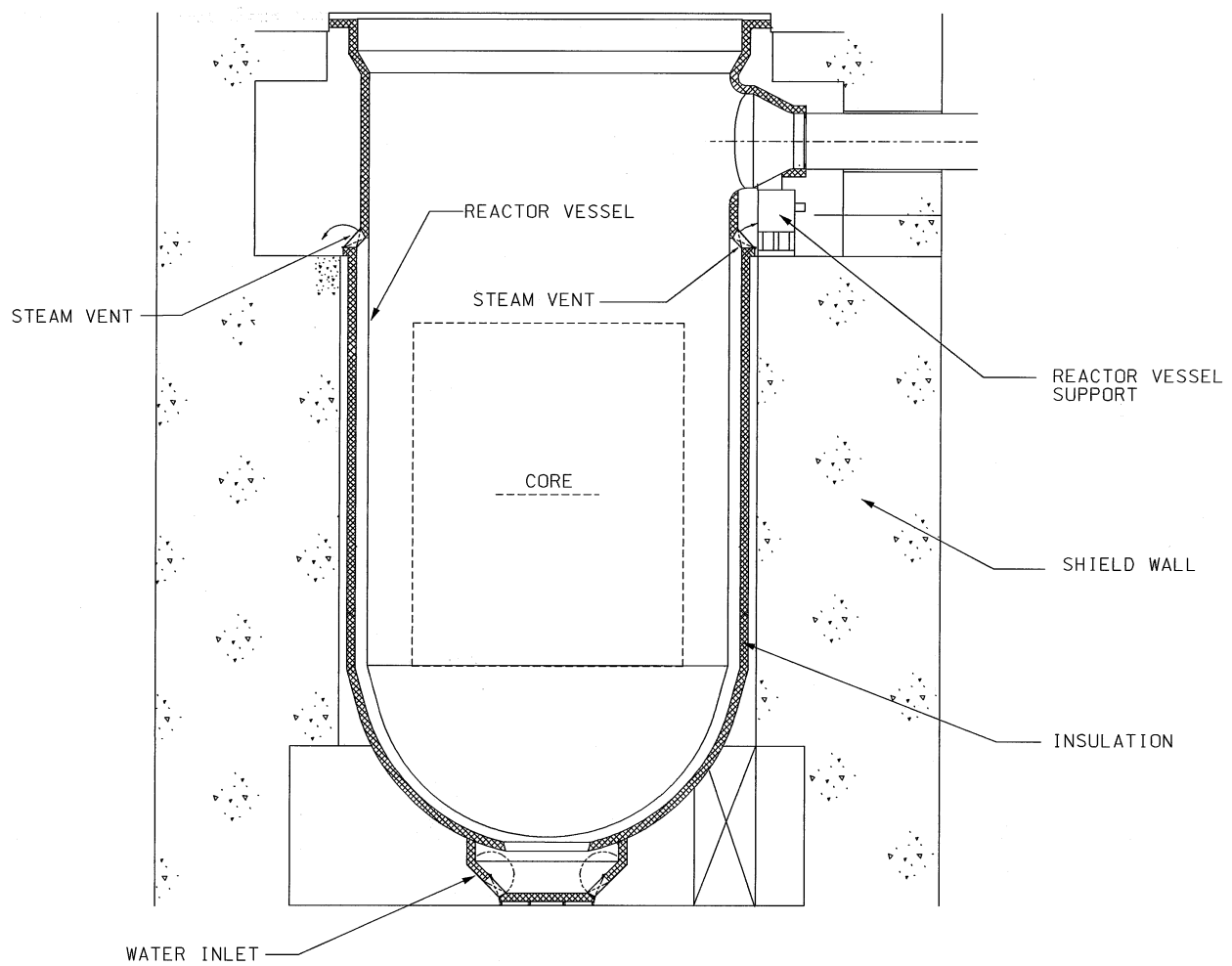SUPPORT

SHIELD WALL

INSULATION

WATER INLET

Figure 5.3-7

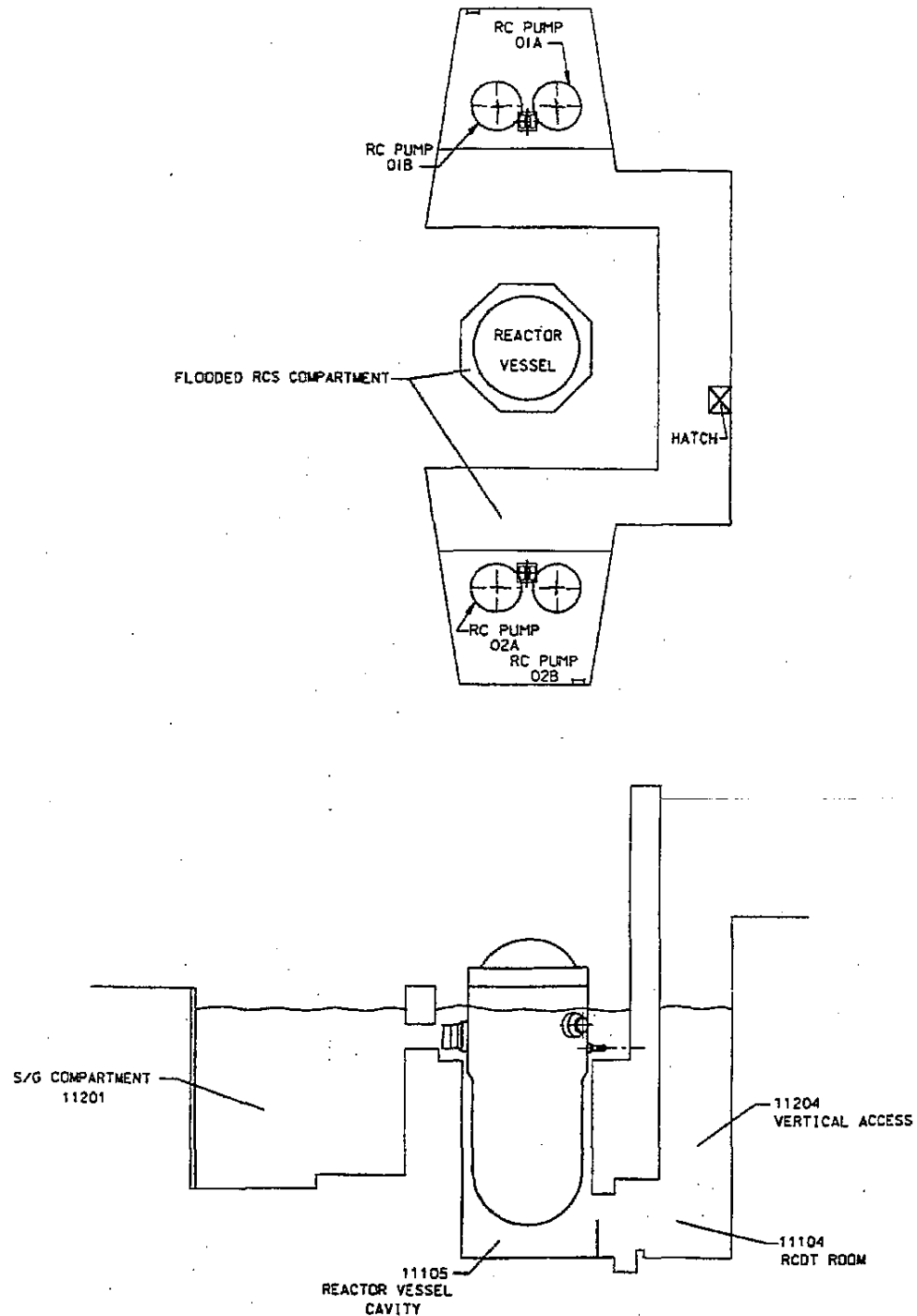**Schematic of Reactor Vessel Insulation**

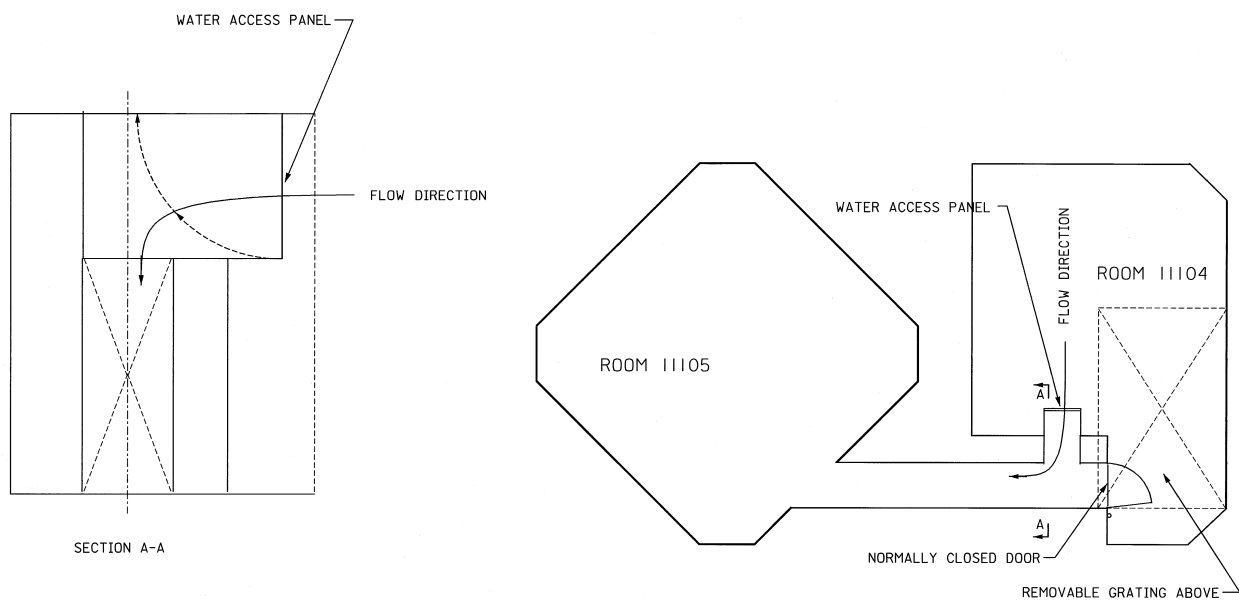Figure 5.3-8

RCS Flooded Compartments During Ex-Vessel Cooling

Figure 5.3-9

Door Between RCDT Room and Reactor Cavity Compartment