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 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.
- 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 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 hemispherical bottom head. The removable flanged hemispherical upper head consists of two sections, the closure head flange and the closure head dome. The closure head flange and dome are fabricated from a low alloy steel forging and plate respectively, 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 welded to the bottom hemispherical head just below the bottom hemispherical head-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 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 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 ring 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 61 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 38 penetrations in the closure head used to provide access for in-core and core exit instrumentation. A tube is inserted into each of the 38 penetrations and is welded to the closure head penetration.

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 AP600 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.

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, Class 3).

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.

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 top instrumentation tubes 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 end-of-life RT_{NDT} and upper shelf energy projections estimated using Regulatory Guide 1.99 for the end-of-life neutron fluence at the 1/4-thickness (T) and ID reactor vessel locations are given in Table 5.3-3.

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. 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.

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

- Dosimeters
  - Iron
  - Copper
  - Nickel
  - Cobalt-aluminum (0.15-percent cobalt)
  - Cobalt-aluminum (cadmium shielded)
  - Uranium-238 (cadmium shielded)
  - Neptunium-237 (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. 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 AP600.

Capsule	Withdrawal Time
1st	When the accumulated neutron fluence of the capsule is 5 x 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 life (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

In order to effect a correlation between fast neutron (E > 1.0 MeV) exposure and the radiation-induced property changes observed in the test specimens, a number of fast neutron flux monitors are included as an integral part of the reactor vessel surveillance program. In particular, the surveillance capsules contain detectors employing the following reactions:

Fe^{54} (n,ρ) Mn^{54}
Ni^{58} (n,ρ) Co^{58}
Cu^{63} (n,α) Co^{60}
Np^{237} (n,f) CS^{137}
U238 (n,f) Cs137

In addition, thermal neutron flux monitors, in the form of bare and cadmium-shielded Co-Al wire, are included within the capsules to enable an assessment of the effects of isotopic burnup on the response of the fast neutron detectors.

The use of activation detectors such as those listed above does not yield a direct measure of the energy dependent neutron flux level at the point of interest. Rather, the activation process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material. An accurate estimate of the average neutron flux level incident on the various detectors 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 operating history of the reactor
- The energy response of the given detector
- The neutron energy spectrum at the detector location

The procedure for the derivation of the fast neutron flux from the results of the Fe^{54} (n,ρ) Mn^{54} reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron energy spectrum, is similar.

The Mn^{54} product of the Fe^{54} (n,ρ) Mn^{54} reaction has a half-life of 314 days and emits gamma rays of 0.84-MeV energy, which are easily detected using a NAI scintillator. In irradiated steel samples, chemical separation of the Mn^{54} may be performed to ensure freedom from interfering activities. This separation is simple and very effective, yielding sources of very pure Mn^{54} activity. In some samples, the interferences may be corrected for by the gamma spectrometric methods without any chemical separation.

The analysis of the sample requires that two procedures be completed. First, the Mn^{54} disintegration rate per unit mass of sample and the iron content of the sample must be measured as described above. Second, the neutron energy spectrum at the detector location must be calculated.

For this analysis, the two-dimensional, multi-group, discrete ordinates transport (DOT) (Reference 2) code is employed to calculate the spectral data at the location of interest. The DOT calculations utilize a 47-group energy scheme and a P3 expansion of the scattering cross sections to compute neutron radiation levels within the geometry of interest. The cross sections used in the analyses are obtained from the SAILOR cross-section library (Reference 3) which was developed specifically for light water reactor applications. The reactor geometry employed here includes a description of the radial regions internal to the primary concrete (core barrel, neutron pad, pressure vessel, and water annuli) as well as the surveillance capsule and an appropriate reactor core fuel loading pattern and power distribution. Thus, distortions in the fission spectrum due to the attenuation of the reactor internals are accounted for in the analytical approach.

Having the measured activity, sample weight, and neutron energy spectrum at the location of interest, the calculation of the threshold flux is as follows.

The induced Mn^{54} activity in the iron flux monitors may be expressed as:

$$D = \frac{No}{A} f_i \int_E \sigma(E)\Phi(E) \sum_{j=1}^{n} F_j\left(1 - e^{-\lambda\tau_j}\right) e^{-\lambda\tau_d}$$

where:

D = induced Mn^{54} activity (dps/g_{Fe})

No = Avogadro's number (atoms/g-atom)

A = atomic weight of iron (g/g-atom)

f_j = weight fraction of Fe^{54} in the detector

$\sigma(E)$ = energy dependent activation cross section for the Fe^{54} (n,ρ) Mn^{54} reaction (barns)

$\phi(E)$ = energy dependent neutron flux at the detector at full reactor power (n/cm^2-s)

λ = decay constant of Mn^{54} (1/s)

F_j = fraction of full reactor power during the jth time interval,τ_j

τ_j = length of the j'th irradiation period(s)

τ_d = decay time following the Jth irradiation period(s)

The parameters F_j, τ_j, and τ_d depend on the operating history of the reactor and the delay between capsule removal and sample counting.

The integral term in the above equation may be replaced by the following relation:

$$\int_0^{\infty} \sigma(E)\phi(E) = \bar{\sigma}\,\bar{\phi}_{E_{th}} = \frac{\sum_0^{\infty} \sigma_S(E)\phi_S(E)}{\sum_{E_{th}}^{\infty} \phi_S(E)}\,\bar{\phi}_{E_{th}}$$

where:

σ = effective spectrum average reaction cross-section for neutrons above energy, E_{TH}

ϕE_{TH} = average neutron flux above energy, E_{TH}

$\sigma_S(E)$ = multigroup Fe^{54} (n,ρ) Mn^{54} reaction cross-sections compatible with the DOT energy group structure

$\phi_S(E)$ = multigroup energy spectra at the detector location obtained from the DOT analysis

E_{TH} = threshold energy for damage correlation

Thus:

$$D = \frac{No}{A} f_i \,\bar{\sigma}\,\bar{\phi}_{E_{TH}} \sum_{j=1}^{n} F_j\,(1 - e^{-\lambda\tau_j})e^{-\lambda\tau_d}$$

or, solving for the threshold flux:

$$\bar{\phi}_{E_{TH}} = \frac{D}{\frac{N_o}{A} f_i \, \bar{\sigma} \sum_{j=1}^{n} F_j \left(1 - e^{-\lambda \tau_j}\right) e^{-\lambda \tau_d}}$$

The total fluence above energy E_{TH} is given by:

$$\Phi_{E_T} = \bar{\phi}_{E_{TH}} \sum_{j=1}^{n} F_j \tau_j$$

where:

$\sum_{j=1}^{n} F_j \tau_j$ = the total effective full power seconds of reactor operation up to the time of capsule removal

Because of the relatively long half-life of Mn^{54}, the fluence may be accurately calculated in this manner for irradiation periods up to about 2 years. Beyond this time, the calculated average flux begins to be weighted toward the later stages of irradiation, and some inaccuracies may be introduced. At these longer irradiation times, therefore, more reliance must be placed on Np^{237} and U^{238} fission detectors with their 30-year half-life product (Cs^{137}).

No burnup correction was made to the measured activities, since burnout of the Mn^{54} product is not significant until the thermal flux level is about 10^{14} n/cm^2-s.

The error involved in the measurement of the specific activity of the detector after irradiation is estimated to be ±5 percent.

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

The energy and spatial distribution of neutron flux within the reactor geometry is obtained from the DOT two-dimensional S transport code (Reference 2). The radial and azimuthal distributions are obtained from an R,θ computation, wherein the reactor core as well as the water and steel annuli surrounding the core are modeled explicitly. The axial variations are then obtained from an R,Z DOT calculation using the equivalent cylindrical core concept. The neutron flux at any point in the geometry is then given by:

$$\phi(E,R,\theta,Z) = \phi(E,R,\theta)\ F(Z)$$

where $\phi(E,R,\theta)$ is obtained directly from the R,θ calculation and F(Z) is a normalized function obtained from the R,Z analysis. Core power distributions representative of time-averaged conditions derived from statistical studies of long-term operation of Westinghouse plants are used. These input distributions, which are characteristic of out-in fuel loading patterns (fresh fuel on the periphery), include rod-by-rod spatial variations for peripheral fuel assemblies.

Benchmark testing of these generic or design basis power distributions against surveillance capsule data from Westinghouse plants indicates that this analytical approach yields conservative results, with calculations exceeding measurements from 10 to 25 percent.

Having the calculated neutron flux distributions within the reactor geometry, the exposure of the capsule as well as the lead factor between the capsule and the vessel may be determined as follows.

The neutron flux at the surveillance capsule is given by:

$$\phi_c = \phi(E, R_c, \theta_c, Z_c)$$

and the flux at the location of peak exposure on the pressure vessel inner diameter is:

$$\sigma_{v\text{-max}} = \phi(E, R_{v\text{-max}}, \theta_{v\text{-max}}, Z_{v\text{-max}})$$

The lead factor then becomes:

$$LF = \frac{\phi_c}{\phi_V - \max}$$

Similar expressions can be developed for points within the pressure vessel wall and thus, together with the surveillance program dosimetry, serve to correlate the radiation-induced damage to test specimens with that of the reactor vessel. The lead factor value for AP600 design is approximately 2.5.

One must realize that the lead factors are sensitive to core power distribution. For example, low leakage fuel management may reduce neutron leakage by differing amounts along the periphery of the core. An altered azimuthal distribution of fast neutron flux, compared to the design basis shape, may change the relationship between the flux at a capsule and the peak flux on the vessel. It is prudent to examine the impact of changing core power distributions on lead factors and adjust surveillance capsule withdrawal schedules when necessary.

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. 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 an expected plant design life 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 in accordance with 10 CFR 50, Appendix G and are applicable up to 54 effective full-power years. These curves are shown in Figures 5.3-2 and 5.3-3. 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 hemispherical bottom head and removable flanged hemispherical upper head. The vessel is fabricated by welding together the lower head, 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 is approximately 38 feet long and has an inner diameter at the core region of 157 inches. The total weight of the vessel is approximately 400 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 AP600 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 welded to 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. The flange is designed to interface properly with a multiple stud tensioner device. 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 attached to the head at the elevation of the lower internals lower core support plate. The lower head is welded to the lower shell course with the weld located outside the higher fluence active core region. The lower shell is a ring forging about eight inches thick with an inner diameter of 157 inches. The length of the shell is greater than 144 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-one 4-inch outer diameter penetrations for the control rod drive mechanism housings and thirty-eight, 1.5-inch O.D. penetrations for the incore instrumentation tubes.

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, General Design Criteria 30 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, except for the vessel support surfaces, are painted with a heat-resistant paint 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 painted with heat-resistant paint 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. The fluence of 2×10^{19} n/cm^2 which is the design basis fluence at the vessel inner radius, at 54 EFPY, at the peak location, was used for calculating the RT_{PTS} value. 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, initial RT_{NDT}, and end-of-life RT_{PTS} values 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. 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. The closure head ferritic pressure boundary welds are examined from the outside diameter only.

The vessel design and construction enables inspection in accordance with the ASME Code, Section XI. The reactor vessel inservice inspection program is detailed in the technical specifications.

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 AP600 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.
- A support frame to prevent the insulation panels from breaking free and blocking water from cooling the reactor vessel exterior surface.

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 retained in the reactor vessel through heat removal from the vessel wall accomplished by the water. This is accomplished by providing a minimum flow area of 7.5 ft^2 in the portions of the flow paths required to vent steam. 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 inlet assembles are designed to minimize the pressure drop during ex-vessel cooling to permit water 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 of the reactor cavity. The vertical 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. A nominal gap (with no deflection) of more than twice the minimum gap is provided.

The conical design of the bottom portion of the vessel insulation is constructed of flat panels. This provides a single point of contact with the spherical portion of the vessel in the event that an insulation panel becomes dislodged. This prevents hot spots from developing on the reactor vessel and permits sufficient flow to maintain in-vessel retention. The nominal gap between the conical portion of the insulation and the spherical portion of the reactor vessel is not less than 9 inches.

The structural frame supporting the insulation is designed to withstand the bounding severe accident loads without exceeding deflection criteria. 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 have 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, and due to the relatively large minimum flow area through each water inlet assembly, with an area of at least 7 in^2, the water inlet assemblies are not susceptible to clogging from debris inside containment. This 7 in^2 minimum area is also provided in the steam vent path to minimize the potential for clogging the steam flow path.

Near the top of the lower insulation segment are steam vent dampers. These dampers are normally closed to prevent reactor vessel heat loss, and a small buildup of steam pressure under the insulation will cause them to open to the vent position. The steam vent dampers are passive, self-actuated devices and will operate when steam is generated under the insulation with the cavity filled with water.

Extensive maintenance of the vessel insulation is not normally required. Periodic verification that 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 a 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 cavith 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 presure drop through the opening during ex-vessel cooling.

5.3.5.4 Determination of Forces on Insulation and Support System

The expected forces that may be expected in the reactor cavity region of the AP600 plant during a core damage accident in which the core has relocated to the lower head and the reactor cavity is reflooded have been conservatively established based on data from the ULPU test program (Reference 5). The particular configuration (Configuration III) reviewed closely models the full-scale AP600 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 III experiments in conjunction with the static head of water that would be present in the AP600 is used to estimate forces acting on the rigid sections of insulation. Further evaluation of the forces on the reactor vessel insulation and supports is provided in the AP600 Probabilistic Risk Assessment.

5.3.5.5 Design Evaluation

A structural analysis of the AP600 reactor cavity insulation system demonstrates that it meets the functional requirements discussed above. The analysis encompassed the insulation and support system and included a determination of the stresses in support members, bolts, insulation panels and welds, as well as deflection of support members and insulation panels.

The results of the analyses 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.

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

5.3.6 Combined License Information

5.3.6.1 Pressure-Temperature Limit Curves

The pressure-temp. curves shown in Figures 5.3-2 and 5.3-3 are generic curves for AP600 reactor vessel design, and they are the limiting curves based on copper and nickel material composition. However, for a specific AP600, 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 applicant during procurement of the reactor vessel. As noted in the bases to Technical Specification 3.4.15, use of plant specific curves requires evaluation of the LTOP system. This includes evaluating the setpoint pressure for the RNS relief valve.

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 Reactor Vessel Materials Properties Verification

The Combined License applicant will address 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.

5.3.6.4 Reactor Vessel Insulation

The Combined License applicant will address verification that the reactor vessel insulation is consistent with the design bases established for in-vessel retention.

5.3.7 References

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

2. Soltesz, R. G., et al., "Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Volume 5 -Two-Dimensional Discrete Ordinates Techniques," WANL-PR-(LL)-034, August 1970.

3. SAILOR RSIC Data Library Collection DLC-76, "Coupled, Self-Shielded, 47 Neutron, 20 Gamma-Ray, P3, Cross Section Library for Light Water Reactors."

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

5. Theofanous, T.G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

Table 5.3-1

MAXIMUM LIMITS FOR ELEMENTS OF THE REACTOR VESSEL

Element	Beltline Forging (percent)	As Deposited Weld Metal (percent)
Copper	0.03	0.03
Phosphorus	0.01	0.01
Vanadium	0.05	0.05
Sulfur	0.01	0.01
Nickel	0.85	0.85

Table 5.3-2 (Sheet 1 of 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
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	

Table 5.3-2 (Sheet 2 of 2)

REACTOR VESSEL QUALITY ASSURANCE PROGRAM

	RT[(a)]	UT[(a)]	PT[(a)]	MT[(a)]
Seal ledge				Yes
Head lift lugs				Yes
Core pad welds			Yes	

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

Note:

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 inprocess/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)	RT_{NDT} (°F)	USE (ft-lb) 1/4T	RT_{PTS} (°F)
Beltline Forging	-10	> 75	23	> 50	54
Head	10	N/A	N/A	N/A	N/A
Flange	10	N/A	N/A	N/A	N/A
Weld	10	N/A	N/A	N/A	N/A
Beltline Weld	-20	> 75	24	> 50	56

Note:

The minimum unirradiated upper shelf energy for beltline base metal is for the transverse direction.

Table 5.3-4

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

	Capsules U, V, W, X, Y, and Z		
Material	**Charpy**	**Tensile**	**1/2T-CT**
Limiting forging (long.)	30	4	6
Limiting forging (trans.)	30	5	6

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.)	44-1
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.)	173
Inside diameter at shell (in.)	157
Inlet nozzle inside diameter (in.)	22
Outlet nozzle inside diameter (in.)	31
Clad thickness, minimum (in.)	0.22
Lower head thickness, minimum (in.)	6
Vessel beltline thickness, minimum (in.)	8
Closure head thickness (in.)	6.25

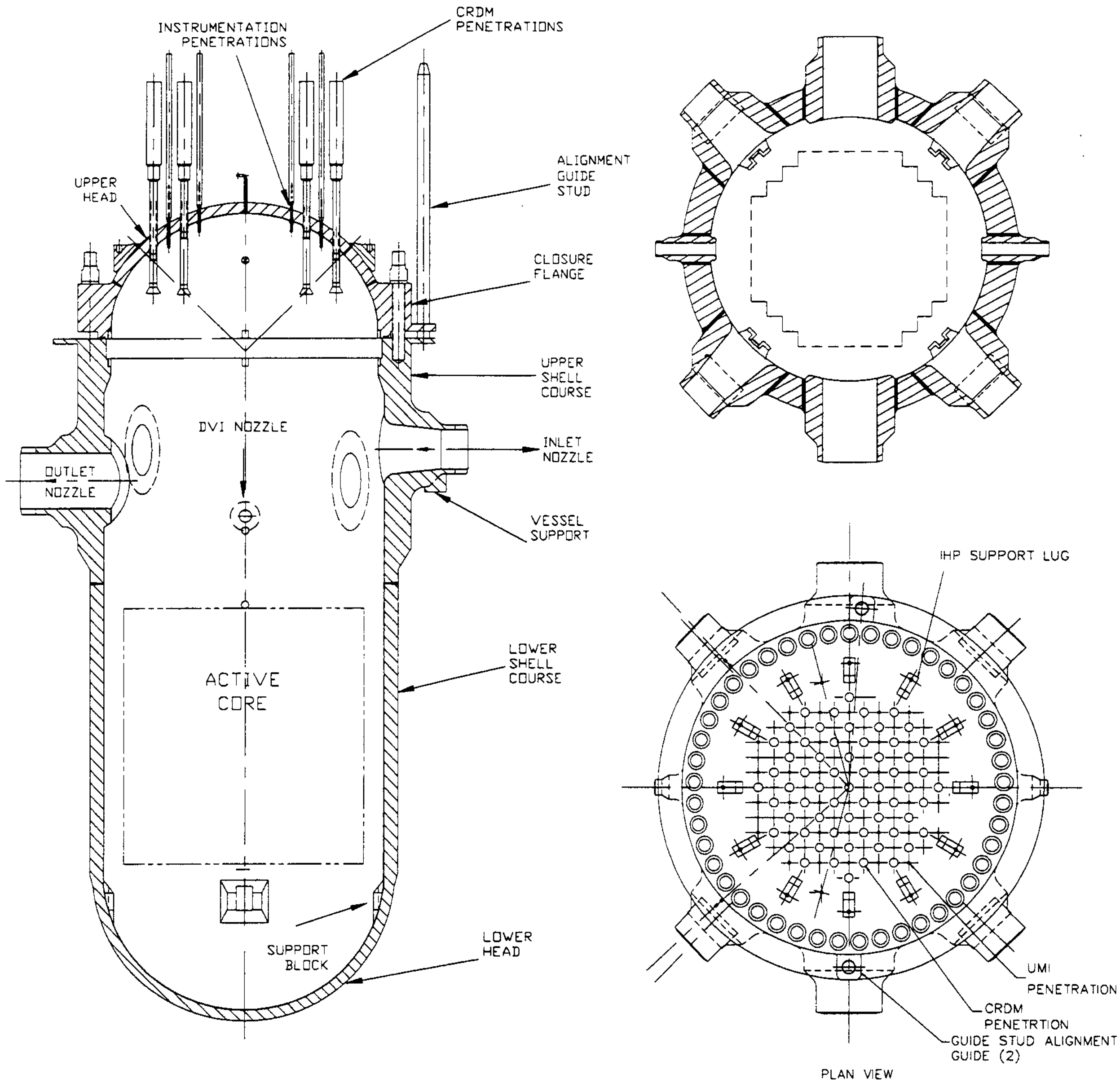


Figure 5.3-1

Reactor Vessel

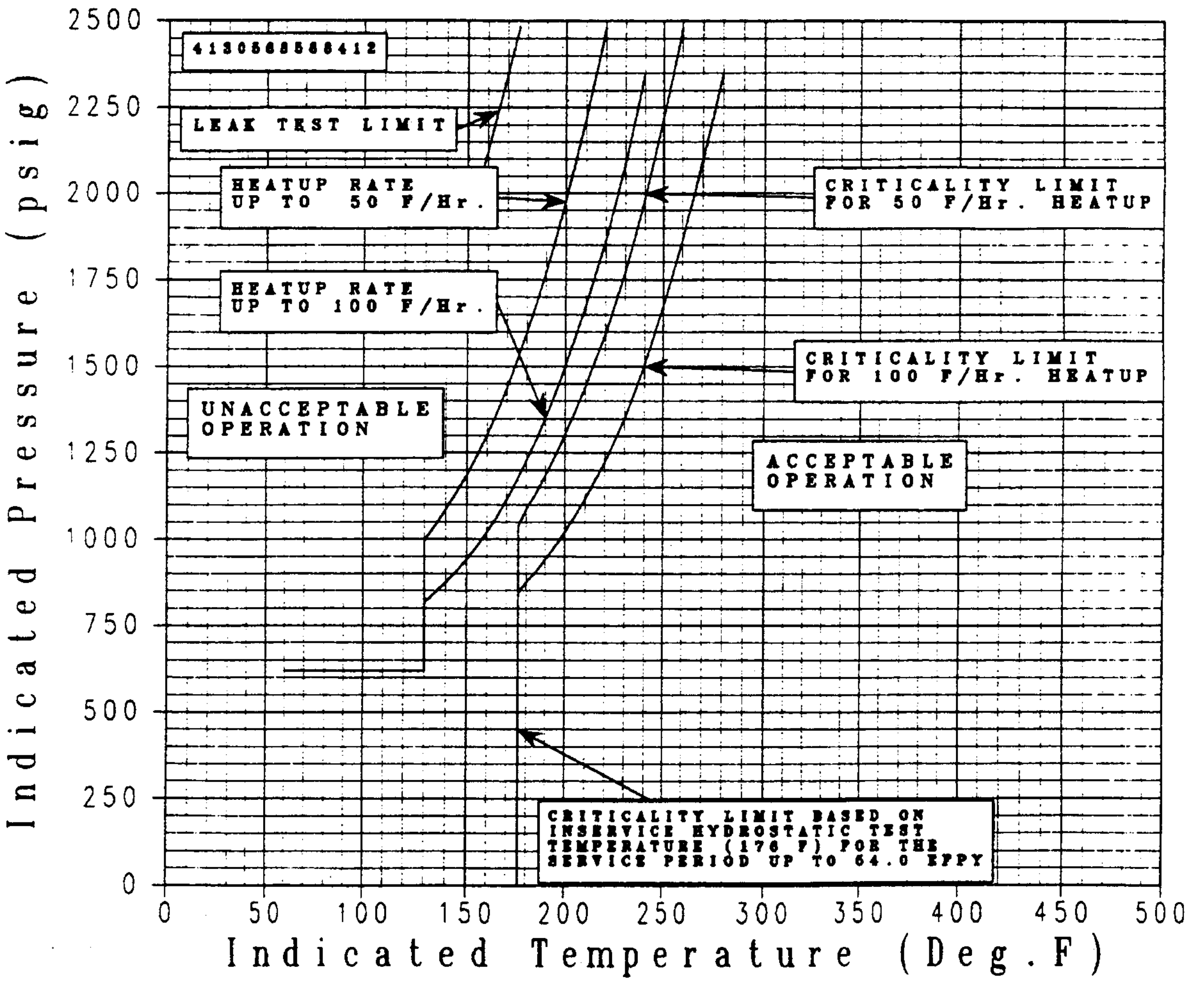


Figure 5.3-2

AP600 Reactor Coolant System Heatup Limitations (Heatup Rate Up to 50 and 100°F/hour) Applicable for the First 54 EFPY (without Margins for Instrumentation Errors)

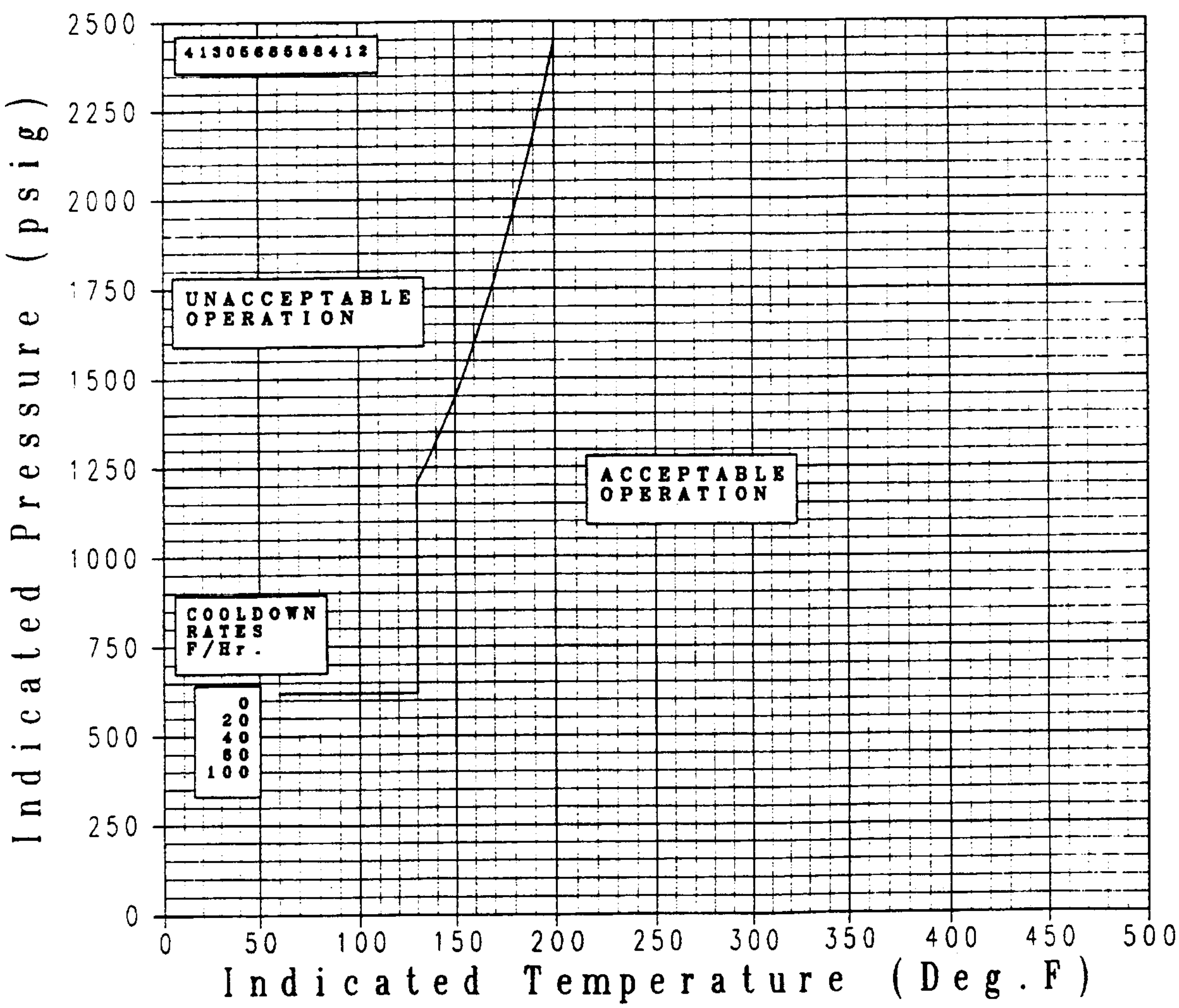


Figure 5.3-3

AP600 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hour) Applicable for the First 54 EFPY (without Margins for Instrumentation Errors)

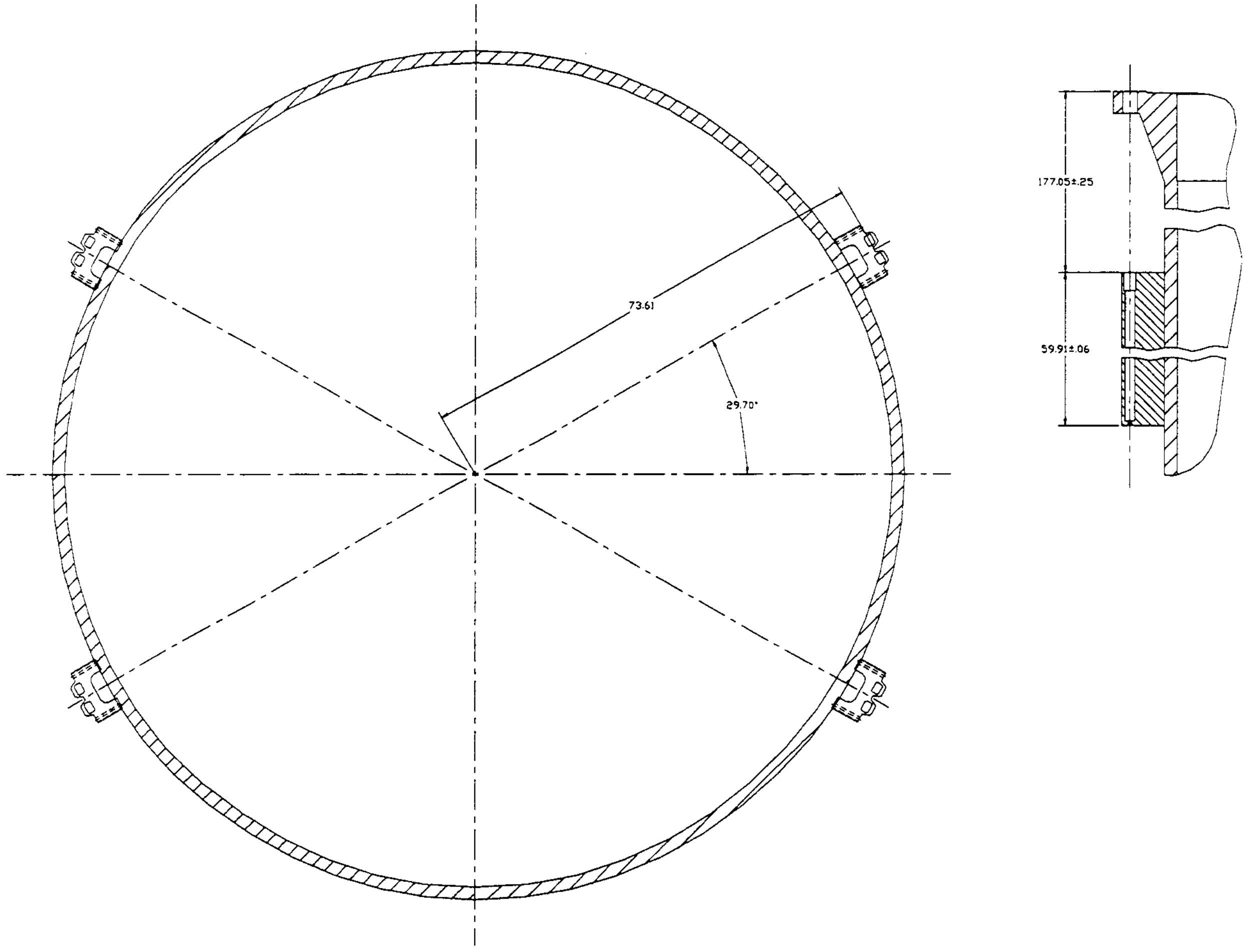


Figure 5.3-4

AP600 Reactor Vessel Surveillance Capsules Locations

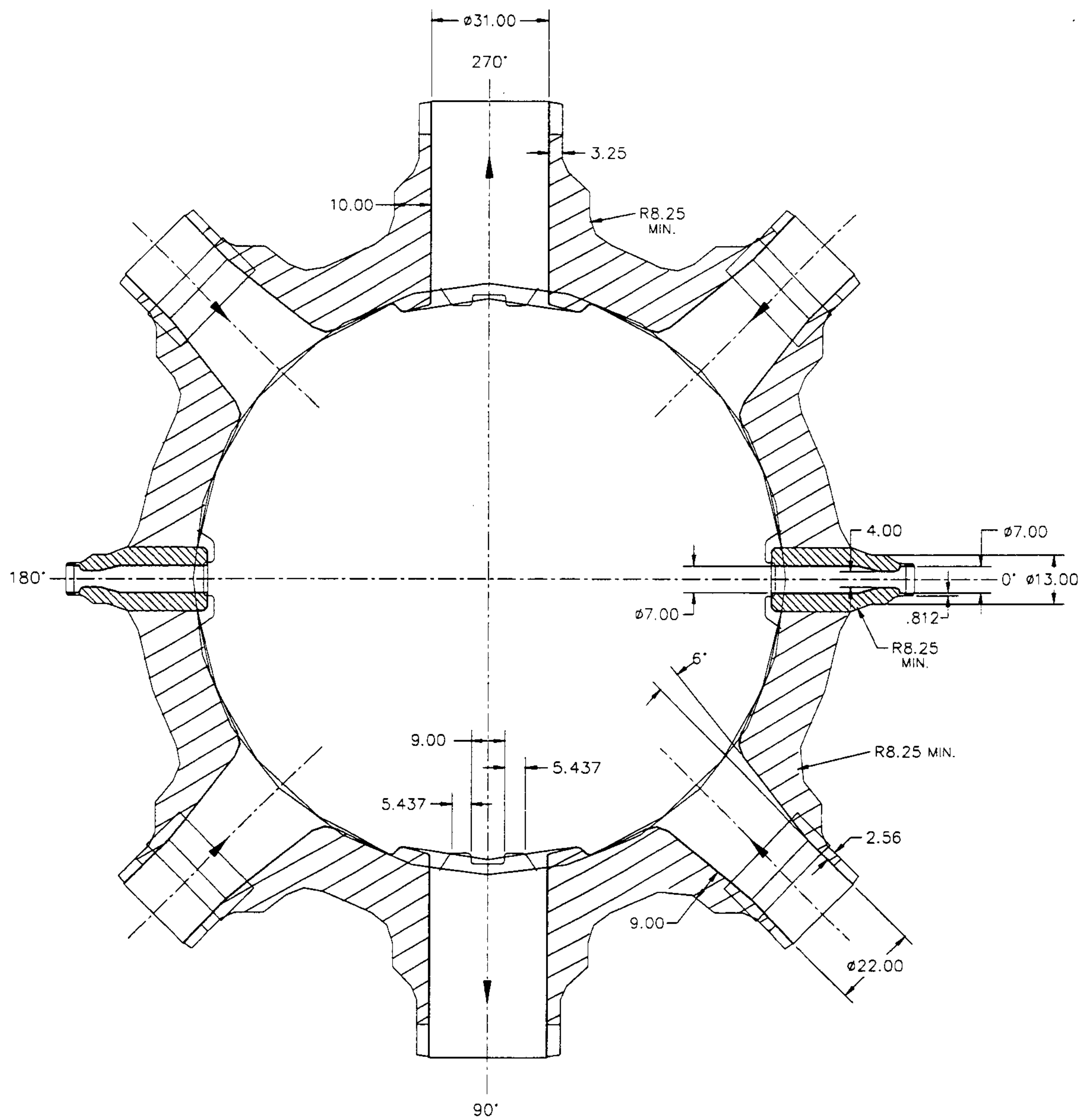


Figure 5.3-5

Reactor Vessel Key Dimensions
Plan View

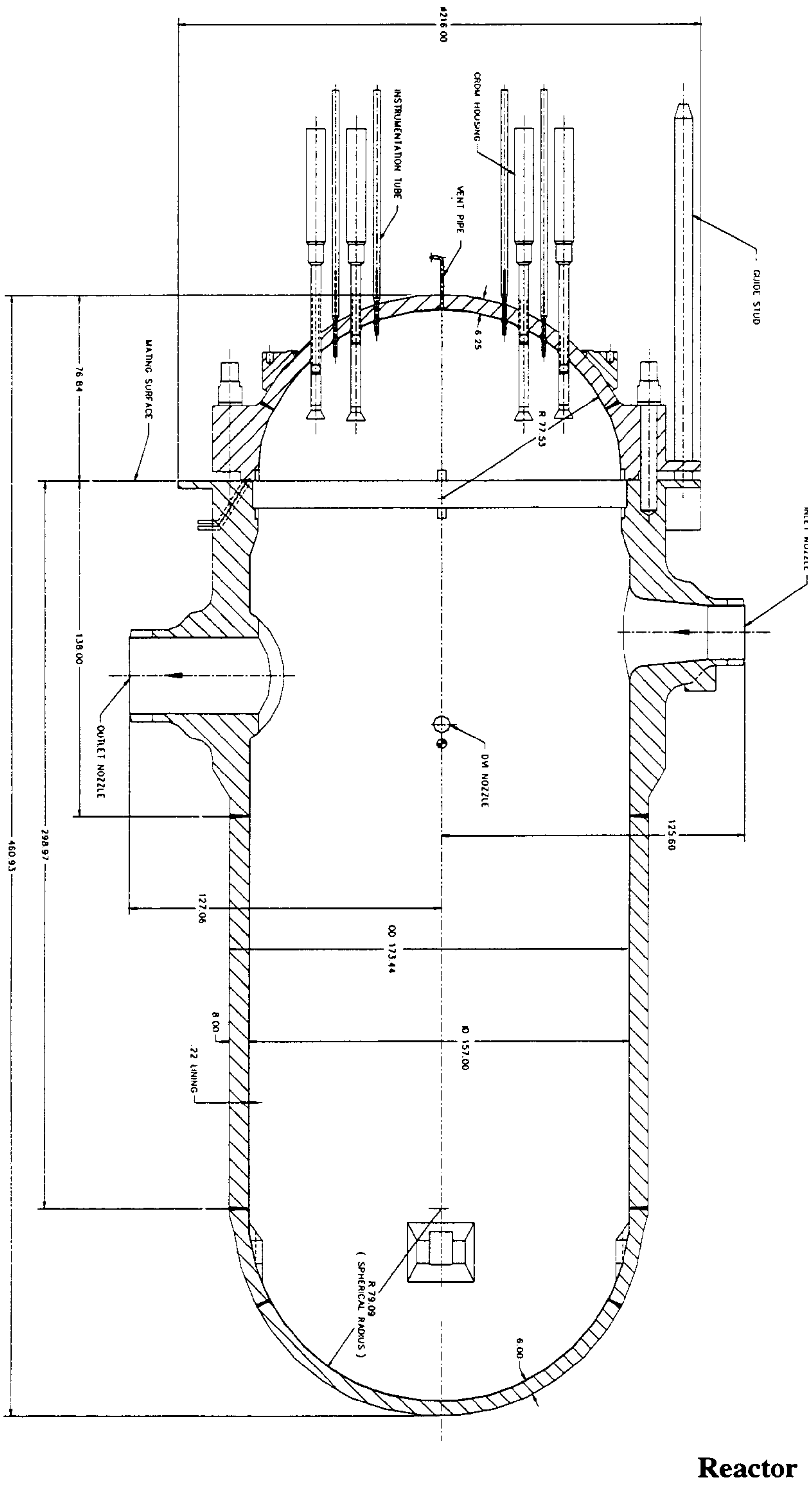
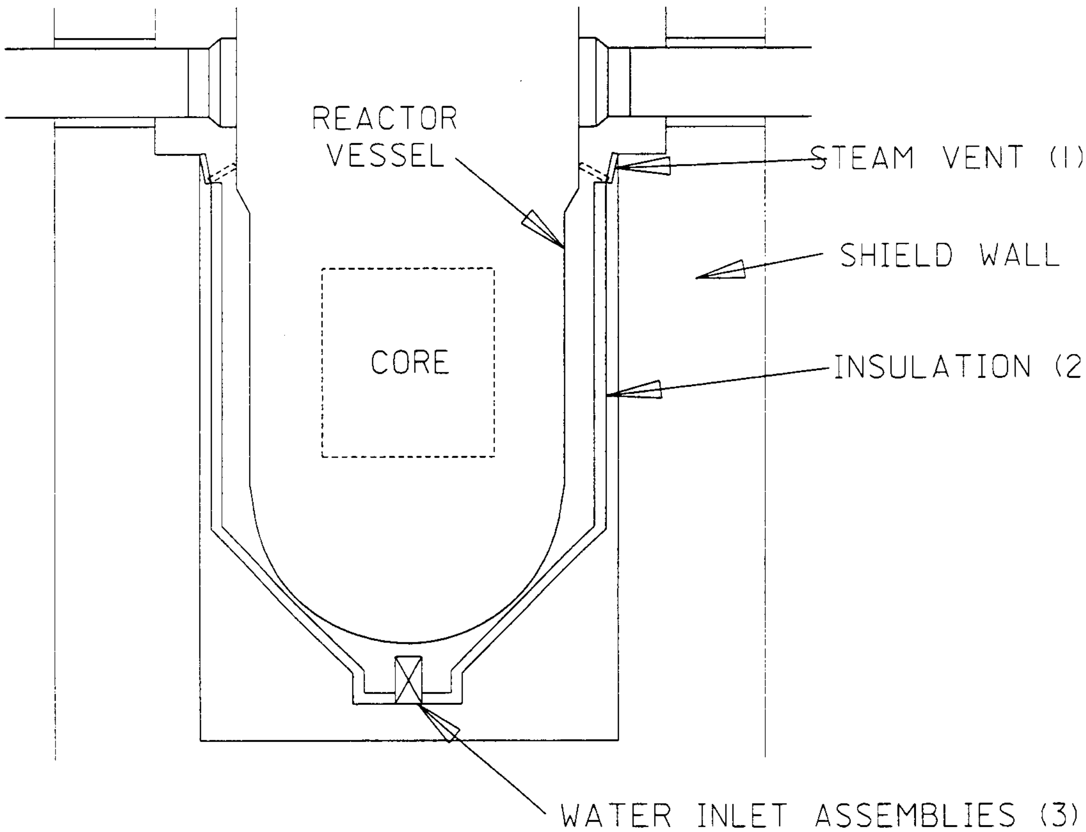


Figure 5.3-6

Reactor Vessel Key Dimensions, Side View



(1) - Minimum steam vent flow area provided in subsection 5.3.5.1
(2) - Minimum gap between insulation and vessel insulation provided in subsection 5.3.5.2
(3) - Minimum flow area provided in subsection 5.3.5.2

Figure 5.3-7

Schematic of Reactor Vessel Insulation

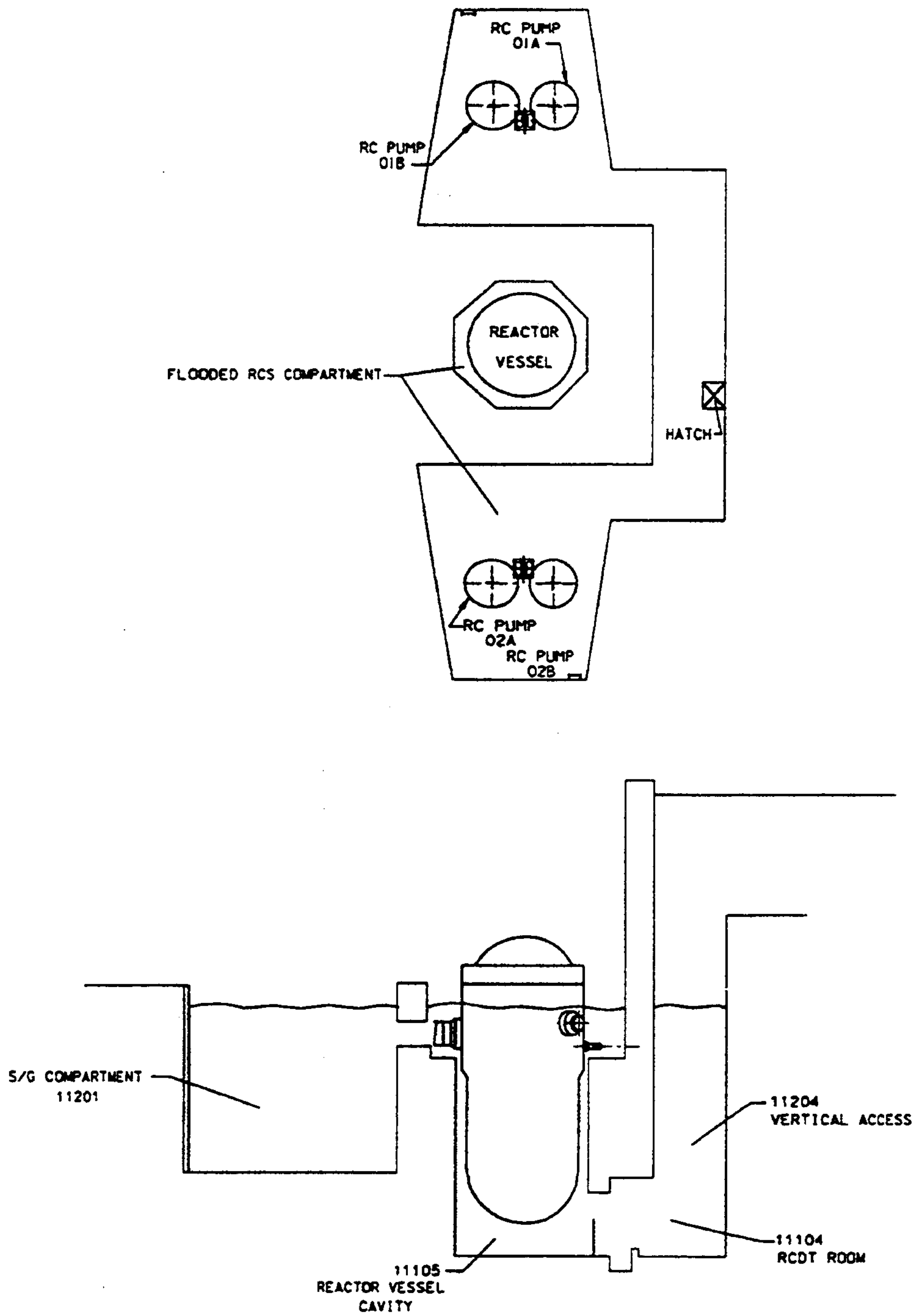


Figure 5.3-8

RCS Flooded Compartments During Ex-Vessel Cooling

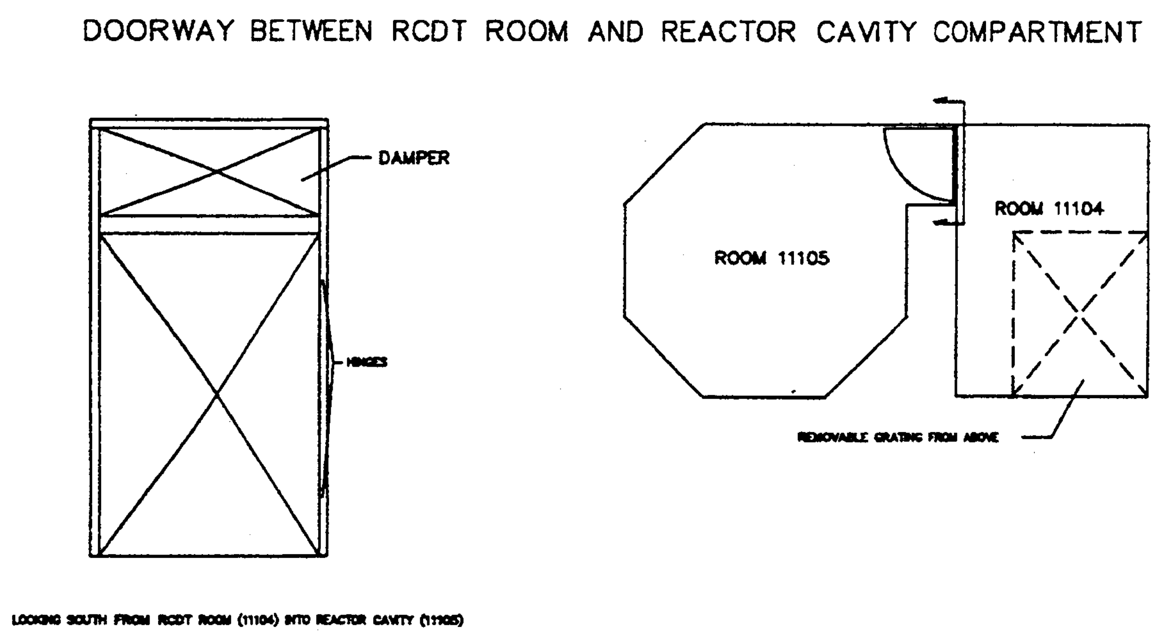


Figure 5.3-9

Door Between RCDT Room and Reactor Cavity Compartment