CHAPTER 4. REACTOR COOLANT SYSTEM

4.1 DESIGN BASES

The Reactor Coolant System circulates coolant at the required flow rate and temperature to meet the heat removal requirements of the core. Heat produced by the nuclear reaction is transferred by the coolant to the steam generators where steam is generated for the turbine plant. The coolant also acts as a neutron moderator and reflector, and as a vehicle for the soluble neutron absorber used in chemical shim control. The Reactor Coolant System also serves safeguard objectives by acting as a barrier against the release of energy and radioactive material into the reactor containment, and by preventing adverse reactivity and thermohydraulic effects resulting from changes in coolant pressure, temperature or soluble absorber concentration.

The system and its auxiliaries are designed to accommodate 10 per cent step changes in plant load and 5 per cent per minute ramp changes over the range from 15 per cent full power up to and including but not exceeding 100% full power without reactor trip. Based on experience, however, greater capabilities are to be expected as the operating conditions will probably not be as pessimistic as those used for the design basis. The Reactor Coolant System will accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass system will accept an instantaneous loss of load of 40 per cent of rated load without reactor trip by steam dump to the condenser.

In designing Reactor Coolant System components, the code requirements listed in Table 4-1 provide adequate standards for materials, design, fabrication, inspection and testing. The design parameters are listed in Table 4-2. Normal operating temperatures and pressures are well below their respective design values, thus providing additional safety margin over the minimum design objectives of the code requirements.

Reactor vessel design is based on the transition temperature method of evaluating the potential for brittle behavior of the vessel material. A design value of 100°F is used as a reasonable estimate of design transition temperature (DTT) for unirradiated material before actual properties are established during fabrication.

TABLE 4-1

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

Component	Codes
Reactor Vessel	ASME III Class A
Steam Generator	
Tube Side	ASME III Class A
Shell Side	ASME III Class C
Pressurizer	ASME III Class A
Pressurizer Relief Tank	ASME III Class C
Pressurizer Safety Valves	ASME III
Reactor Coolant Piping	USA S.I. B31.1
Primary Pump Casing	ASME III Class A

TABLE 4-2

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM

Reactor Heat Output, MWt	3025
Reactor Heat Output, Btu per hour	$10,324 \times 10^6$
Operating Pressure, psig	2235
Reactor Inlet Temperature °F	549.7
Reactor Outlet Temperature °F	607.7
Number of Loops	4
Design Pressure, psig	2485
Design Temperature, °F	650
Hydrostatic Test Pressure (cold), psig	3110
Coolant Volume, including pressurizer volume, cu. ft.	12,200
Total Reactor Flow, gpm	354,000

For design purposes a shift in the nil ductility transition temperature (NDTT) of 275F° is selected as a value which will assure plant operation, heatup and cooldown without undue operating limitations over the plant design lifetime. Experience in calculating the operating limits of reactors previously analyzed[(1),(2)] indicates that a significant margin exists between this assigned limit and prescribed plant startup and cooldown procedures. In addition, the design time-integrated fast neutron flux of 3.7×10^{19} n/cm^2 (energies > 1 mev) obtained from present reference irradiation curves (See Fig. 4-1) for the 275F° shift also leaves an ample margin over the calculated vessel irradiation exposure of 1.4×10^{19} n/cm^2 for the plant design life (Based upon a power level of 3217 MWt).

4.2 REACTOR COOLANT SYSTEM DESIGN AND FABRICATION TECHNIQUES

Extensive consideration has been given to the requirements of design, fabrication and inspection to produce highest quality components for use in the reactor coolant system, assuring adequate conservatism and full use of practical existing inspection techniques.

4.2.1 DESIGN

The rules of Section III Nuclear Vessels (ASME B&PV Code) provide an up-to-date industry-wide acceptable basis for design evaluation of nuclear vessels. The criteria established by Section III are used for evaluating the design of the reactor vessel, pressurizer, coolant pump casing and steam generator.

(1) U.S. AEC Docket No. 50-206 San Onofre Reactor Vessel Design Report

(2) U.S. AEC Docket No. 50-213 Connecticut-Yankee Facility Description and Safety Analysis Report, Chapter 5.1

The ground rules established for fatigue analysis, unique to Section III of the boiler codes are based on low cycle fatigue considerations and cover discontinuities, stress raisers, and thermal as well as pressure and mechanically induced stresses. The conservatism of the fatigue design curves in Section III has been verified by the cyclic testing of pressure vessels carried out by Southwest Research Institute for the AEC and the Pressure Vessel Research Council.

The reactor coolant piping is analyzed in accordance with the requirements of USA S.I. Code for Pressure Piping. While this procedure does not categorize all stresses (primarily the expansion stress which is peculiar to piping) in the same manner as Section III, it does provide a basis for fatigue analysis which has been correlated with strain cycling of piping subassemblies by Markl of **Tube Turns, Inc.** This fatigue analysis takes into account the interrelation of the primary system components, piping and supports and the restrained thermal growth.

The design specifications include all of the loads the reactor vessel, pressurizer, pump casing and steam generator will carry, which include static and dynamic loads from internal or external sources, steady state, operational and abnormal transient conditions expected during the life of the system. The hydraulic forces imposed by the instantaneous failure of reactor coolant piping are considered in the design. The operational modes of the plant are based on a conservative evaluation of nuclear plant operation coupled with experience from existing nuclear plants such as Yankee-Rowe and include startup and shutdown cycles, load step changes, reactor trips and pre-startup hydrostatic tests.

The design pressure of the reactor coolant system components provides a 10% margin above the steady state operating pressure, providing a significant range to cover transients without exceeding the setting of the safety valves.

A complete stress analysis which reflects consideration of all design loadings detailed in the design specification is prepared by the manufacturer to assure compliance with the stress limits of Section III for the reactor vessel, steam generator, pressurizer and pump casing. Westinghouse independently will review these stress analyses. A similar analysis of the piping will be prepared by Westinghouse or for Westinghouse by a qualified piping analysis contractor.

As part of the design control on materials, and in addition to that reported for the reactor vessel, Charpy V-notch tests are run on all ferritic materials used in fabricating pressure parts of the steam generator and pressurizer to assure hydrotesting and operation in the ductile region at all times.

4.2.2 INSPECTION DURING FABRICATION

The degree of conservatism considered in the quality assurance of materials and fabrication procedures is indicated in the attached Table 4-3 on page 4-8 which delineates all the inspection requirements that are imposed by Westinghouse on its equipment suppliers. In addition to the inspections shown are those the equipment supplier performs to confirm the adequacy of material he receives, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator are governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication are governed by USA S.I. B31.1, Westinghouse and Consolidated Edison requirements and are equivalent to those performed on ASME coded vessels.

Specifications and procedures for performing the examinations are consistent with those established in Section III of the ASME B&PV Code and are reviewed by Westinghouse and Consolidated Edison. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material is fabricated, orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the plate surface most subject to damage as a result of the heat treating, rolling, forging, forming and fabricating processes, receives a 100% surface inspection by Magnetic Particle or Liquid Penetrant Testing after all these operations are completed. Although flaws in plates are inherently laminations in the center, all reactor coolant plate material is subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. (All forgings receive the same inspection.) In addition, 100% of the material volume is covered in these tests as an added assurance over the grid basis required in the code.

Westinghouse and Consolidated Edison or its agents' Quality Control engineers will monitor the supplier's work, witnessing key inspections not only in the supplier's shop but at sub-vendors on the major forgings and plate material. Their normal surveillance includes verification of records of material, physical and chemical properties, review of radiographs, performance of required tests and qualifications of supplier personnel.

4.2.3 FABRICATION

The equipment specifications require that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse. These specifications are then reviewed by qualified Westinghouse engineers. This also is done on the field fabrication procedures to assure that installation welds are of equal quality.

Section III of the ASME Boiler and Pressure Vessel Code requires that nozzles carrying significant externals loads shall be attached to the shell by full penetration weld. This requirement will be carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop are made using full penetration welds.

Preheat requirements, non-mandatory under Code rules, are performed on all weldments, including P1 and P3 materials which are the materials of construction in the reactor vessel, pressurizer and steam generators. Preheat and post-heat of weldments both serve a common purpose - the production of tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones whereas post-heating achieves this by tempering any hard zones, which may have formed due to rapid cooling. Reactor coolant system components utilize both preheat and post-heat.

TABLE 4-3

Reactor Coolant System
Quality Assurance Program

Component	RT*	UT*	PT*	MT*	ET*
1. Steam Generator					
1.1 Tube Sheet					
1.1.1 Forging		yes		yes	
1.1.2 Cladding		yes	yes		
1.2 Channel Head					
1.2.1 Casting	yes			yes	
1.2.2 Cladding			yes		
1.3 Secondary Shell & Head					
1.3.1 Plates		yes		yes	
1.4 Tubes		yes			yes
1.5 Nozzles (forgings)		yes		yes	
1.6 Weldments					
1.6.1 Shell, longitudinal	yes			yes	
1.6.2 Shell, circumferential	yes			yes	
1.6.3 Cladding		yes	yes		
1.6.4 Nozzle to shell	yes			yes	
1.6.5 Support brackets				yes	
1.6.6 Tube-to-tube sheet			yes		
1.6.7 Instrument connections				yes	
1.6.8 Temporary attachments after removal				yes	
1.6.9 After hydrotest (all welds)				yes	
2. Pressurizer					
2.1 Heads					
2.1.1 Casting	yes			yes	
2.1.2 Clad			yes		
2.2 Shell					
2.2.1 Plates		yes		yes	
2.2.2 Clad			yes		
2.3 Heaters					
2.3.1 Tubing		yes	yes		
2.3.2 Centering of element	yes				
2.4 Nozzle		yes	yes		
2.5 Weldments					
2.5.1 Shell, longitudinal	yes			yes	
2.5.2 Shell, circumferential	yes			yes	
2.5.3 Cladding			yes		
2.5.4 Nozzles	yes			yes	
2.5.5 Nozzle Safe Ends	yes			yes	

*RT - Radiographic Testing
UT - Ultrasonic Testing
PT - Dye Penetrant Testing
MT - Magnetic Particle Testing
ET - Eddy Current Testing

TABLE 4-3 (Continued)

Component	RT	UT	PT	MT	ET
3. Piping					
3.1 Fittings (Castings)	yes		yes		
3.2 Fittings (Forgings)		yes	yes		
3.3 Pipe		yes	yes		
3.4 Weldments					
3.4.1 Longitudinal	yes		yes		
3.4.2 Circumferential	yes		yes		
3.4.3 Nozzle to run pipe	yes		yes		
3.4.4 Instrument connections			yes		
4. Pumps					
4.1 Casting	yes		yes		
4.2 Forgings		yes	yes		
4.3 Weldments					
4.3.1 Circumferential	yes		yes		
4.3.2 Instrument connections			yes		
5. Reactor Vessel					
5.1 Forgings					
5.1.1 Flanges		yes		yes	
5.1.2 Studs		yes		yes	
5.1.3 Head Adapters		yes		yes	
5.2 Plates		yes		yes	
5.3 Weldments					
5.1.1 Main Seam	yes			yes	
5.1.2 CRD Head Adapter Connection			yes		
5.1.3 Instrumentation Tube			yes		
5.1.4 Main Nozzles	yes	yes		yes	
5.1.5 Cladding			yes		
5.1.6 Nozzle Safe Ends	yes		yes	yes	

4.3 SYSTEM OPERATION

The Reactor Coolant System consists of four closed loops connected to a reactor vessel. Each loop contains a steam generator, a pump, piping, and instrumentation. A pressurizer is connected to the hot leg of one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system is shown in Figure 4-2*.

Pressure in the system is controlled by the pressurizer, where water and steam are maintained in equilibrium to minimize pressure variations due to contraction and expansion of the coolant. Steam can either be formed by electrical heaters, or condensed by a controlled water spray. Spring-loaded steam safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the steam is condensed and cooled by mixing with water.

All materials are corrosion resistant in the coolant operating environment. The Chemical and Volume Control System maintains reactor coolant chemistry in accordance with the composition of Table 4-4. Periodic analyses are made during operation to check the coolant composition.

TABLE 4-4

REACTOR COOLANT WATER CHEMICAL COMPOSITION

Oxygen, ppm, max.	0.1
Chloride, ppm, max.	0.15
Fluoride, ppm, max.	0.1
Hydrogen, cc (STP) per kg H_2O	25-35
Total suspended solids, ppm operating max.	1.0
pH control agent	0.3×10^{-4} to 3.2×10^{-4} molal strong base alkali
Boron as boric acid	0 to 4000 ppm B

* The containment boundary shown on the process flow diagram indicates those major components which are to be located inside the reactor containment. The intersection of a process line with this boundary indicates a functional penetration; however, the actual number of piping penetrations required will be determined in the final designs.

4.4 COMPONENTS

4.4.1 REACTOR VESSEL

The reactor vessel is a vertical cylinder with a hemispherical bottom and a flanged and gasketed upper head. Figure 4-3 is a schematic of the vessel, which is constructed of SA 302 Grade B, low alloy steel. All internal surfaces are clad with corrosion resistant stainless steel.

Coolant enters through four nozzles in the side of the reactor vessel and flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction. After passing upward through the reactor core, the coolant flows out of the vessel through four exit nozzles located on the same level as the inlet nozzles.

A one piece thermal shield, concentric with the reactor core, is attached to the outside of the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attenuating much of the gamma radiation and some of the fast neutrons that escape from the core. This shield minimizes thermal stresses in the vessel resulting from heat generated by gamma energy. The weight of the thermal shield is supported by lugs at the bottom of the core barrel. The thermal shield is stabilized at the top by guide lugs mounted on the core barrel. (See Figure 3-35, Chapter 3.)

The vessel and its internals are constructed so as to permit removal of the internals if ever required during plant life. The closure head is bolted to the vessel and is removed for refueling. A double gasketed seal with a monitored leakoff is provided for sealing the head.

The requirements of the ASME Boiler and Pressure Vessel Code, Section III, for Class A Nuclear Pressure Vessels provide quality standards which are supplemented in specific areas by additional Westinghouse requirements. These requirements call for additional material tests and inspections resulting in a more comprehensive knowledge of the material, which in turn provides a better evaluation of the reactor vessel design for the anticipated operating modes.

Mechanical loads considered in the design analysis of the reactor vessel supports include the dead weight of the vessel, head, insulation, control rod drive mechanisms, head-lifting rig, ventilation shroud, thermal shield, core, core support structure, contained coolant, and the partial weight of the reactor coolant piping bearing on the vessel. To this load are added the dynamic forces of the control rods following trip, and the reaction forces of the reactor coolant loops induced by thermal deflection. The reactor vessel is a Class I seismic component and the design of the supports considers seismic loadings.

Loads included in the design analysis of the internal supporting ledge of the vessel include the weight of the thermal shield, core, and core support structure, the dynamic forces of the control rods following trip and the clamping force exerted on the core support by the head closure.

The operational mode of the plant is based on realistic evaluation of the nuclear plant behavior coupled with experience from existing nuclear plants such as the Yankee-Rowe, Selni and others. Table 4-5 summarizes the reactor vessel design parameters and design transients.

TABLE 4-5

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL

Design pressure, psig	2485
Design temperature, °F	650
Operating pressure, psig	2235
Inside diameter of shell, in.	173
Outside diameter across nozzles, in.	254
Overall height of vessel and enclosure head, ft.-in.	42 -4

Vessel Design Transients	Number of Cycles*
a. Plant heatup at 100°F per hour	200
b. Plant cooldown at 100°F per hour	200
c. Plant loading at 5% of full power per minute	14,500
d. Plant unloading at 5% of full power per minute	14,500
e. Step load increase 10% of full power but not to exceed full power	2,000
f. Step load decrease of 10% of full power	2,000
g. Step load decrease of 50% of full power	200
h. Reactor trip and attendant temperature transients	400
i. Hydrostatic Test (3110 psig)	5

The entire vessel design, material ordering drawings and material specifications will be carefully reviewed by Westinghouse to determine that it complies with the requirements of the specification prior to material procurement and start of fabrication. Supplier fabrication process, and nondestructive test specifications are also reviewed by Westinghouse for conformance to codes and Westinghouse requirements. The stress analysis report prepared by the vessel manufacturer and certified by his registered engineer

* Some of the above transient conditions are conservative estimates for equipment design purposes and are not intended to be accurate representations of actual transients or to reflect actual operating procedures.

specialized in analytical stress analysis is independently reviewed by Westinghouse personnel with extensive background in the design and analysis of reactor vessels.

Fabrication of the vessel is carefully planned by the manufacturer and includes a step by step record of all fabrication and inspection efforts which demonstrate compliance with specifications and standards. These efforts include the control of steel making for plate and forgings, their testing properties initially and after heat treatment inspection and includes compensation for stress relieving operation. All material is stamped with a coded identification number to preclude any doubt as to origin or destination. These records are regularly reviewed by Westinghouse engineering and quality control representatives who are specialized in pressure vessel design engineering, materials engineering and quality control with extensive experience in the reactor vessel field and associated engineering efforts. Particular attention is given to vessel areas where operating conditions have the most significant effects on vessel performance and analysis.

For example, the inlet and outlet nozzles are one-piece forgings which are made to fine grain practice and ultrasonically inspected to code standards. Also, the transition between the closure flanges and the upper vessel course and the head dome respectively are given special consideration by selecting plate with superior properties for these applications.

Westinghouse requires a 100% volumetric ultrasonic inspection of the reactor vessel materials. Mapping of both forgings and plate materials used in the reactor vessel is performed.

The detection of flaws during fabrication of the reactor vessel will be accomplished by the following non-destructive testing techniques.

1. Radiographic Examination - all pressure containing welds.

2. Ultrasonic Examination - all plates, forgings, closure studs, pipes tubes and nozzle welds.

3. Magnetic Particle Examination - all surfaces to be clad or welded. All unclad surfaces and welds after final heat treatment. Closure stud surfaces before and after threading.

4. Liquid Penetrant Examination - all cladding after vessel stress relief.

The acceptance standards for all above inspections are consistent with Section III of the ASME Code.

The ground rules used in establishing the cyclic capability of the vessel are those defined in Section III of the ASME Code. These design criteria have been verified in the cycling tests on pressure vessels conducted by Southwest Research Institute and financed jointly by AEC and the Pressure Vessel Research Committee. The evaluation of these cycling tests against the fatigue design curves of Section III of the ASME Code demonstrates the conservatism used in establishing the fatigue design curves.

Westinghouse participates in and actively follows the PVRC programs on fatigue studies which involve studies on propagation of flaws during vessel life. The Westinghouse Atomic Power Divisions and the Westinghouse Research Laboratory are currently seeking to establish programs on crack growth which will evaluate the variables of defect size and shape, orientation, stress levels including cycling profiles and fracture toughness data. These programs are aimed at developing linear elastic fracture mechanics and crack propagation phenomena for the prevention of brittle type pressure vessel failure. The fracture mechanics specimens inserted in the reactor vessel surveillance program can provide data on radiation effects which would be used in the establishing interrelationships of the application of fracture mechanics to crack growth phenomena.

Reactor vessels are fabricated from low alloy steel of plate and forging quality. These steels are medium-strength steels made of fine grain practice, quenched and tempered which enhances the notch toughness, which will insure a low initial NDTT (nil ductility transition temperature).

In the core region of the reactor vessel it is expected that the notch toughness of the material will change as a result of fast neutron exposure. This change is evidenced as a shift in the NDTT and factored into the plant operating procedures in such a manner that full operating pressure is not obtained until the affected vessel material is above the now higher NDTT and in the ductile material region. The partial pressure during startup and shutdown at the temperature below NDTT is maintained below the threshold of concern for safe operation.

Material properties have been obtained in previous reactors both from the reactor vessel vendors test data and the information from the Westinghouse reactor vessel surveillance program which is used as the base reference point from which measurement of changes in the nil ductility transition temperature are determined. In addition, samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing. Measurement of the nil ductility transition temperature properties of the core region plates is performed in accordance with ASTM E185 (Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors). This recommended practice is based primarily on the monitoring of tensile and impact tests. Westinghouse has expanded the surveillance program to include not only the conventional tensile and impact tests, but also fracture mechanics type specimens. The fracture mechanics specimens are the Wedge Opening Loading (WOL) specimens with which Westinghouse and other Westinghouse divisions have had most success with the characteristic low strength, high toughness SA302 Grade B material. The observed shifts in transition temperature with radiation of the core region materials will be used to confirm the calculated limits to startup and shutdown transients.

Reactor Vessel Irradiation Sample Surveillance Program

In the surveillance programs, the evaluation of the radiation damage is based on pre- and post-irradiation testing of Charpy V-notch and tensile specimens. Wedge opening loading (WOL) test specimens provide an alternate backup phase of evaluation. These programs are 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, and are in accordance with ASTM E185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors".

The reactor vessel surveillance programs utilize eight specimen capsules which are located about 3 inches from the vessel wall directly opposite the center portion of the core.

The capsules can be removed and replaced when the vessel head is removed. The capsules contain reactor vessel steel specimens from the shell plates located in the core region of the reactor and associated weld metal and heat affected zone metal. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through Subcommittee II of ASTM Committee E10 Radioisotopes and Radiation Effects are inserted in the capsules. The eight capsules will contain approximately 64 tensile specimens, 384 Charpy V-notch specimens (which will include weld metal and heat affected zone material) and 96 WOL specimens. Dosimeters including pure Ni, Al-Co, (0.15%), Cd shielded Al-Cl, Cd shielded Np-237 and Cd shielded U-238 are placed in impact specimens, tensile specimens or filler blocks drilled to contain the dosimeters. The dosimeters will permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion.

The tentative schedule for removal of capsules is as follows:

Capsule	Estimated Exposure Time
1	Replacement of 1st region
2	Replacement of 2nd region
3	Replacement of 4th region
4	10 years
5	20 years
6	30 years
7	40 years
8	Extra capsule for complementary testing

Irradiation of the specimens will be higher than the irradiation of the vessel because the specimens are located in the vicinity of the core corners and are closer to the core than the vessel itself. Since these specimens will experience higher irradiation and they are actual samples from the materials used in the vessel, the NDTT from the specimens will be conservatively representative of the vessel at a later time in life.

Full Charpy V-notch impact transition curves are obtained for pressure containing forgings and plates used in the reactor vessel shell courses. Westinghouse reviews the procedures and specifications used in the fulfillment of these requirements as to the locations of test specimens and the thermal and fabrication history of the specimens. A concurrent review of the attained properties is made with the vessel vendor.

The application of fracture mechanics to the low strength, high toughness reactor vessel materials is becoming an increasingly useful technique. It is expected that the fracture mechanics techniques will in the future provide quantitative criteria for stress limits that will supplement the pressure transition temperature criteria for setting operating limits.

The vessel interior is protected from direct exposure to the reactor coolant water by a stainless steel cladding. This cladding is applied by a weld overlay process which is superior to other cladding methods for this application.

In summary, the reactor vessel will be designed and a stress analysis performed by the latest techniques and this design and stress analysis will receive a thorough independent review by experienced Westinghouse personnel. The inspection requirements of the ASME Nuclear Vessel Code are augmented by Westinghouse requirements and Westinghouse in-shop inspections. Extensive records of material properties are developed and retained for use in evaluation of future vessel conditions and operation. The closure head and vessel are operated and monitored under most favorable conditions for early detection of leakage. The vessel is available for inspection during service life and vessel material samples are placed in the reactor to permit an accurate evaluation of changes in NDTT because of the effects of neutron exposure.

4.4.2 STEAM GENERATORS

Each loop contains a vertical shell and U-tube steam generator. A typical steam generator of this type is shown in Figure 4-4. Principal design parameters are listed in Table 4-5. The reactor coolant enters the inlet channel head at the bottom of the steam generator through a nozzle, flows through the U-tubes to an outlet channel and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition. Manways are provided to permit access to both the inlet and outlet of the primary side of the tube plate.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly. The vanes impart a radial acceleration to the mixture and separate the water particles from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another passage through the steam generator.

The steam rises through additional separators to limit the moisture content of the steam to one fourth of one per cent or less under all steady load conditions. A manway is provided for access to the moisture separating equipment.

The steam generator is constructed primarily of low alloy steel. The heat transfer tubes are Inconel. The interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel, and the side of the tube sheet in contact with the reactor coolant is clad with Inconel.

As indicated, the primary side of the steam generator is defined according to the ASME code, Section III as a Class A vessel and the secondary side is defined as a Class C vessel. As with the reactor vessel, certain of the ASME code inspection requirements are augmented by Westinghouse requirements and in-shop inspections. For example, the Inconel clad tube sheet is subjected to both ultrasonic and dye-penetrant tests. Also, the secondary shell and head plates are ultrasonically tested and magnetic particle tested. Also, all weldments both on primary and secondary sides (longitudinal, circumferential and nozzles) are subjected to magnetic particle tests.

TABLE 4-6

PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS

Number of Units	4
Type	Vertical U-tube with integral-moisture separator
Tube Side Design Pressure, psig	2485
Tube Side Design Temperature, °F	650
Tube Side Design Flow, lb/hr	33.28×10^6
Shell Side Design Pressure, psig	1085
Shell Side Design Temperature, °F	600
Operating Pressure, Tube Side, Nominal, psig	2235
Operating Pressure, Shell Side, Maximum psig (No Load)	1005
Maximum Moisture at Outlet at Full Load, %	1/4

4.4.3 REACTOR COOLANT PUMPS

Each loop contains a vertical single stage centrifugal pump of the controlled leakage type. A view of a controlled leakage pump is shown in Figure 4-5 and principal design parameters for the pumps are listed in Table 4-7.

The coolant is drawn up through the bottom of the impeller, discharged through spiral passages in the diffuser and out through an exit in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pump in contact with the reactor coolant are constructed of corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, as well as a second seal which directs the controlled leakage out of the pump and a third seal which minimizes the leakage of vapor from the pump into the containment atmosphere.

High pressure water is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the shaft, through the controlled leakage seal, leaves the pump and returns to the seal water injection circuit. A very small amount which leaks through the secondary seal is also collected and piped to the Waste Disposal System.

Component cooling water is supplied to the bearing coolers and the thermal barrier cooling coil by the Auxiliary Coolant System.

The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated radial bearing provides support for the pump shaft.

Design analysis of the pump pressure parts are in accordance with Article 4 of ASME Boiler and Pressure Vessel Code, Section III. Fabrication and inspection of the pump casing is covered by USA S.I. B31.1 Code for Pressure Piping, Nuclear Case N-10, and the additional inspection techniques covered in Section 4.2.2 also apply to the casing.

TABLE 4-7

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS

Number of Units	4
Type	Vertical, single stage radial flow with bottom suction and horizontal discharge
Design Pressure, psig	2485
Design Temperature, °F	650
Operating Pressure, nominal, psig	2235
Suction Temperature, °F (Normal)	**549.7**
Design capacity, gpm	88,500
Design Head, ft.	277
Motor Type	A-C Induction, single speed
Motor Rating	6000 HP

4.4.4 PRESSURIZER

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure.

Under normal full power operating conditions, water fills about one-half of the pressurizer. The pressurizer vessel contains replaceable, direct immersion heaters and is equipped with multiple safety and relief valves, a spray nozzle, interconnecting piping, valves and instrumentation. The electric heaters, located in the lower section of the vessel, are capable of raising the temperature of the pressurizer and its contents at the desired rate during startup of the reactor plant.

The pressurizer is designed to accommodate positive and negative pressure surges caused by load transients. During a positive surge (caused by a decrease in plant load), power operated spray valves admit water from the cold leg of a coolant loop to condense steam in the pressurizer so that the pressure will not increase to a value that would actuate power operated relief valves. In addition, the spray valves can be operated manually by a switch in the central control room. A small continuous spray flow prevents excessive cooling of the spray piping. The resultant recirculation of reactor coolant through the pressurizer minimizes the difference between reactor coolant loop and pressurizer boron concentrations.

Power-operated relief valves and code safety valves attached to the top of the pressurizer protect the reactor coolant system against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray.

During a negative surge (caused by an increase in plant load), flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is constructed of low alloy steel and internal surfaces are clad with weld deposited stainless steel. The heaters are sheathed in austenitic stainless steel. Principal design parameters of the pressurizer are given in Table 4-8.

TABLE 4-8

PRINCIPAL DESIGN PARAMETERS FOR THE PRESSURIZER AND PRESSURIZER RELIEF TANK

<u>Pressurizer</u>	
Design pressure, psig	2485
Design temperature, °F	680
Operating pressure, nominal, psig	2235
Approximate internal volume, cu. ft	1800
Type of Heaters	Direct Immersion

TABLE 4-8 (Continued)

Pressurizer Relief Tank	
Design pressure, psig	100
Design temperature, °F	340
Normal water temperature	120
Total volume, cu. ft.	1600

Inspections performed on the pressurizer, a Class A vessel under the definitions of ASME Boiler and Pressure Vessel Code, Section III, include dye penetrant inspection of the stainless steel clad on the head castings, 100% volumetric ultrasonic test and magnetic particle testing of the shell plates, dye penetrant testing of the heater tubing and radiographic examination and magnetic particle testing of the weldments (longitudinal, circumferential and nozzles).

4.4.5 PRESSURIZER RELIEF TANK

Steam discharged from the pressurizer relief and safety valves passes to the pressurizer relief tank which is partially filled with water at or near ambient containment conditions. The cool water condenses the discharged steam and the condensate is drained to the Waste Disposal System. The tank normally contains water covered by a predominantly nitrogen atmosphere. Steam is discharged under the water level to condense and cool by mixing with the water. The tank is equipped with a spray and drain which are operated to re-establish normal conditions in the tank following a discharge. The tank is protected against a steam discharge that would exceed the design pressure by a rupture disc which discharges into the reactor containment. The tank is carbon steel with a corrosion resistant coating on the internal surface.

4.4.6 PIPING

The reactor coolant piping and fittings which make up the loops are 29 in. I.D. in the hot legs, 27-1/2 in. I.D. in the cold legs and 31 in. I.D. between the steam generator and the reactor coolant pump inlet. All surfaces exposed to the reactor coolant fluid are austenitic stainless steel. The design pressure is 2485 psig and design temperature is 650°F.

The pressurizer relief line which connects the pressurizer safety and relief valves to the discharge nozzle flange on the pressurizer relief tank is carbon steel. Smaller piping, including the pressurizer surge and spray lines, drains, and connections to other systems, is austenitic stainless steel.

Piping is fabricated in accordance with USA S.I. B31.1 Code for Pressure Piping and all special nuclear rulings and supplemented by additional design, inspection and quality requirements which provide a level of quality consistent with that for Class A vessels under the ASME Boiler and Pressure Vessel Code.

4.4.7 VALVES

All valve surfaces in contact with reactor coolant are austenitic stainless steel or a material of equivalent corrosion resistance.

Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System. Double packing and leakoff connections are provided on other valves if required by valve size and fluid conditions.

The power-operated relief valves and spring loaded safety valves are connected to the steam space of the pressurizer. A locally-adjusted throttling valve in parallel with the surge spray valves permits a small continuous circulation spray flow through the spray line and the pressurizer.

4.5 IN-SERVICE INSPECTION CAPABILITY

With regard to the reactor coolant system components, the layout of the equipment and support structures will be designed to permit access to the following areas for examination during a plant shutdown. Access implies ability to visually examine surfaces.

1) The reactor vessel will be available for inspection in the following areas:

   a) Closure studs
   b) Vessel closure head - inside and outside and flange area. Inspection of the inside of the head is done by remote viewing.
   c) Vessel internal surface - the upper portion is available for remote viewing during refueling. The internals can be removed completely to permit inspection of the entire internal surface, if deemed necessary.
   d) Vessel nozzle to pipe weld - design of the shielding will permit access to the nozzle to pipe weld area.

2) The reactor coolant piping and fittings external to the primary shield surrounding the reactor vessel will be available for external surface examination.

3) The pressurizer will be available for external surface and volumetric examination. Internal surface examination is possible.

4) The steam generator shell will be completely available for external surface and volumetric examination. Surface examination of the inside surface of the steam drum is possible.

5) The steam generator channel head will be completely available for surface and volumetric examination. Access to the inside surface is possible.

6) The external surfaces of the pump casing are available for surface examination. With removal of the pump, the internal surfaces are available for inspection.

These areas will be subjected to periodic in-service inspection at frequencies which will be established prior to initial operation of the plant. At the operating license stage, a definitive inspection program will be submitted.

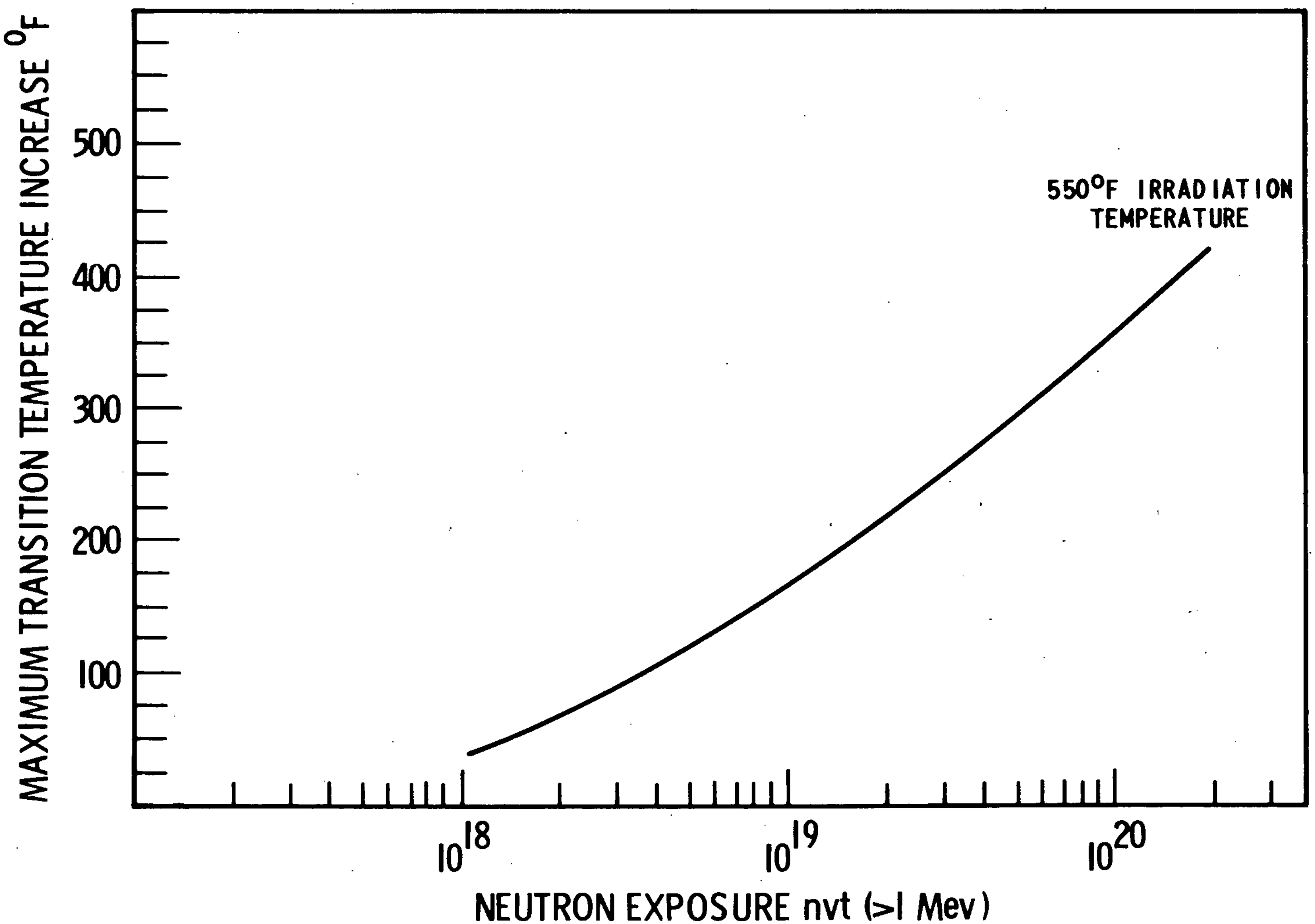
4.6 IN-SERVICE REACTOR COOLANT SYSTEM LEAKAGE DETECTION

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by one or more of the following:

a) Leakage through the head to vessel closure joint will result in a flow to the leak-off provided between the double gaskets of the closure joint which will show up as a high temperature in this line.

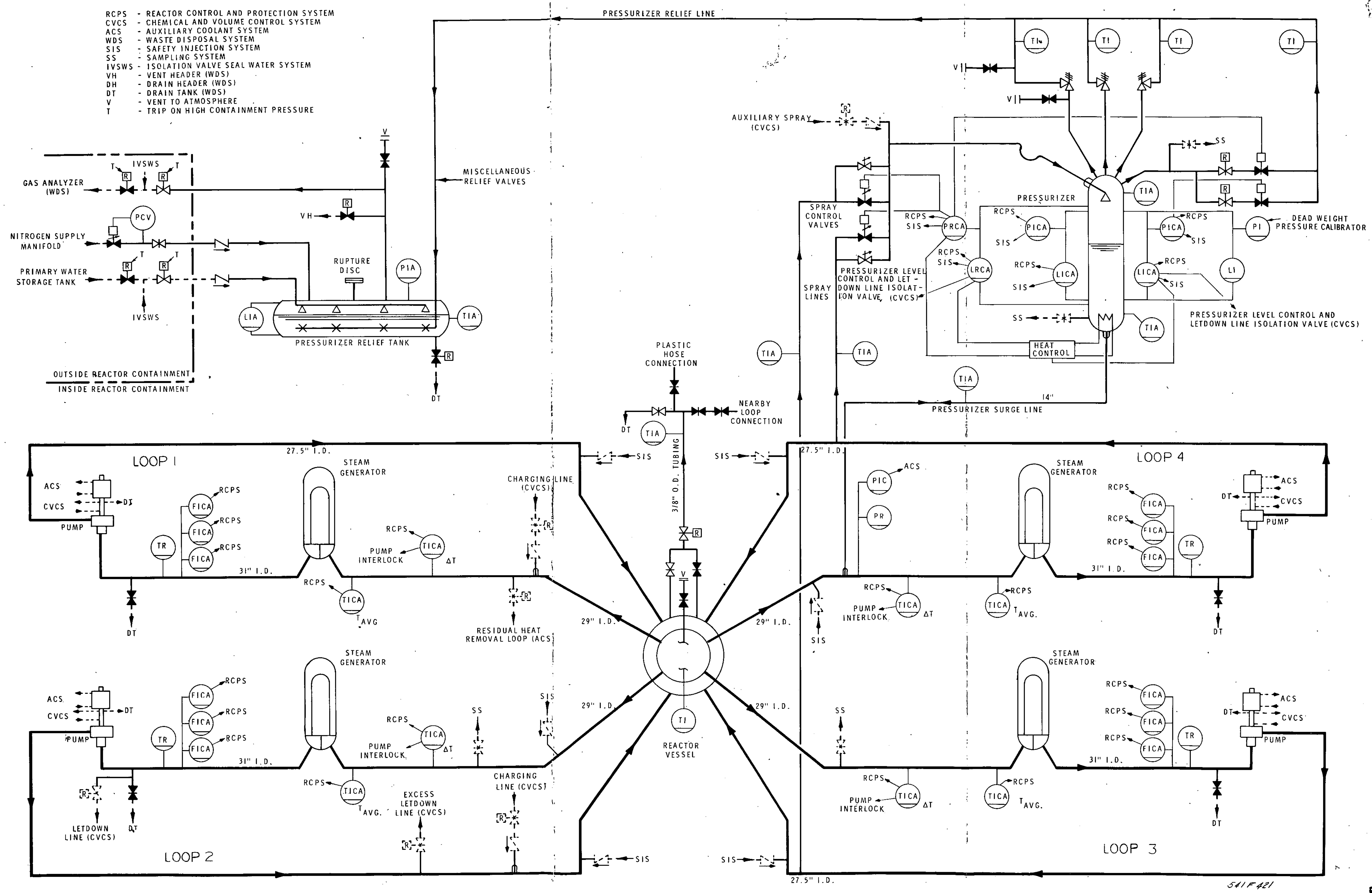
b) Any leakage will cause an increase in the amount of make up water required to maintain a normal level in the pressurizer.

c) The most sensitive indication of reactor coolant system leakage is the containment air particulate monitoring system. Experience has shown that the particulate activity in the atmosphere responds very rapidly to increased leakage. A system will be provided to monitor particulate activity from the areas enclosing the reactor coolant system components so that any leakage from them will be easily detected.

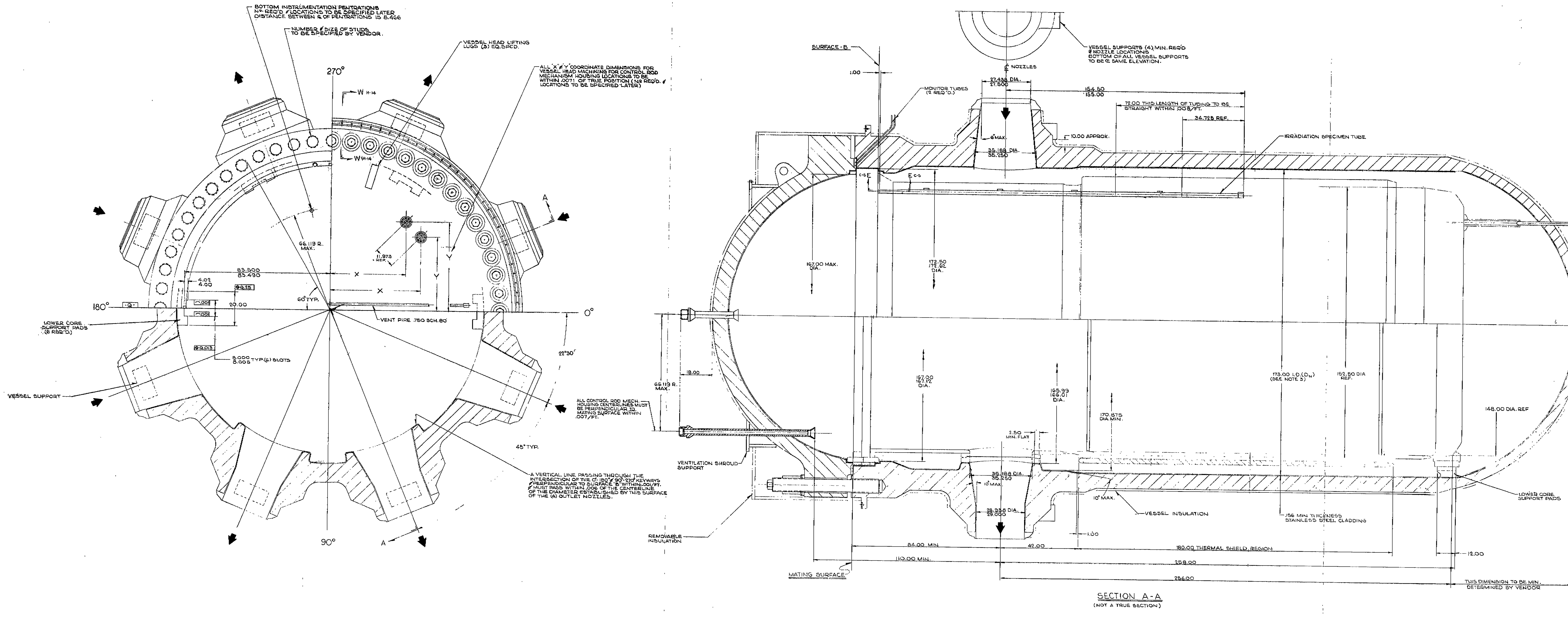


RADIATION EFFECTS ON PRESSURE VESSEL STEEL

RADIATION EFFECTS ON PRESSURE VESSEL STEEL
FIG. 4-1

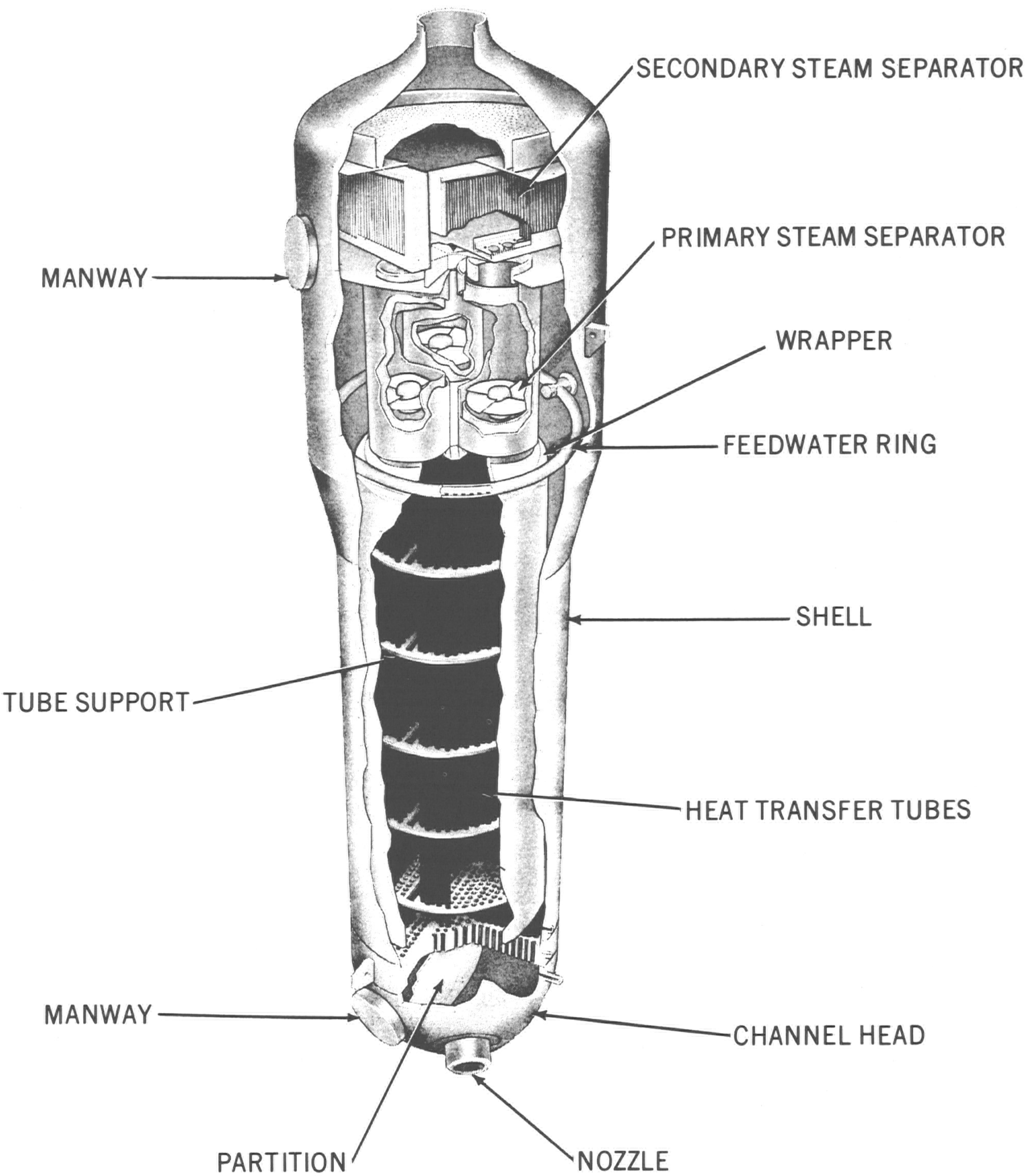


REACTOR COOLANT SYSTEM
FIG. 4-2

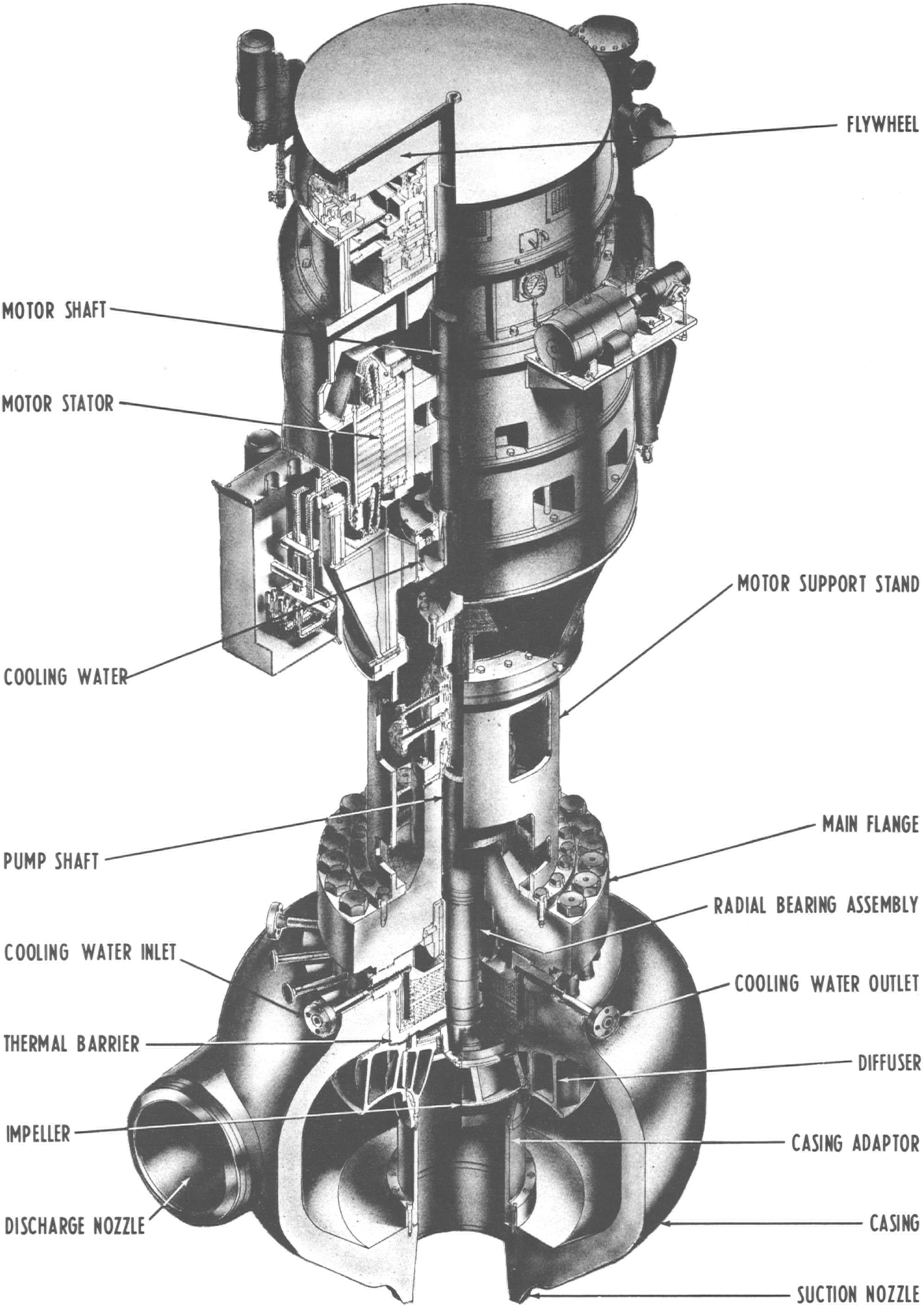


REACTOR VESSEL OUTLINE
FIG. 4-3

STEAM GENERATOR



STEAM GENERATOR
FIG. 4-4



REACTOR COOLANT PUMP

REACTOR COOLANT PUMP
FIG. 4-5

SECTION 5

PSAR Section	Page	Remarks
5.1	5-1	Information on the containment system is expanded in detail in various supplements to the PSAR. Supplements 2 and 4 are almost exclusively devoted to this topic. Specific references to material in other supplements are given below.
5.1.1	5-1	Item 1 of Supplement 1 to the PSAR gives the general design criteria for the design bases described in Section 5.1.1. The following GDC are specifically referenced: GDC 37 page 48 GDC 49 page 64 GDC 50 page 65 GDC 54 page 69 GDC 55 page 70 GDC 56 page 71 GDC 57 page 72
5.1.2.2	5-6	Information on the tornado design criteria and loading of the containment is given in Supplement 1, Item 6; Supplement 4, Question 2 (1), (2), (3); Supplement 5, Item 15; and Supplement 7, Question 2 to add to Section 5.1.2.2 (f).
5.1.2.4	5-9	Additional information on the insulation and its properties are given in Supplement 4, Question 2.12 to add to item (d) of Section 5.1.2.4.
5.1.2.5	5-14	Information on tornado missile criteria is given in Supplement 1, Item 5; information on turbine generator missiles is given in Supplement 1, Item 16 (E-4.5) and Supplement 4, Question 2.2; consideration of the main coolant pump fly wheel as a possible internal missile is given in Supplement 7, Question 3.
5.1.2.6	5.15	The quality control program information of Section 5.1.2.6 has been expanded in Item 5 of Supplement 1, Question 6 of Supplment 2, and in Item 4 of Supplement 5.
5.1.3.8	5-22	Seismic Design criteria for structures and equipment are given in Supplement 1, Item 15. Seismic stress limit curves are given in Item 13 of Supplement 5.

SECTION 5

PSAR Section	Page	Remarks
5.1.3.8	5-23	The table of damping factors has been superseded as given in Supplement 2, Question 2.7 (d).
5.1.3.8	5-23	The general analytical model for the containment seismic response described in Section 5.1.3.8 is superseded by Supplement 2, Question 2.7 (a).
Figure 5-1		The typical reinforcing at the juncture of the cylindrical shell and nut has been revised as shown in Figure 2.4 (a) - 3 of Supplement 2.
Figure 5-10		The typical piping penetrations as shown in Figure 5-10 have been revised as shown in Figure 2.10 (a) - 2 of Supplement 4.