NT

SECTION 4

REACTOR COOLANT SYSTEM

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The following figures are controlled drawings and can be viewed and printed from the applicable listed aperture card.

4.3 COMPONENT AND SYSTEM DESIGN AND OPERATION

4.3.1 General Description

All components of the reactor coolant system are located within the containment building. A flow diagram of the system is shown in P&ID E-23866-210-110. The system includes two heat transfer loops connected in parallel to the reactor vessel. Each loop contains one steam generator, two reactor coolant pumps, flow and temperature instrumentation, and connecting piping. A pressurizer is connected to one of the reactor vessel outlet (hot leg) pipes by a surge line. The pressurizer is located with its base at a higher elevation than the reactor vessel piping. This eliminates the need for a separate drain on the pressurizer, and ensures that it is drained before maintenance. The equipment arrangement relative to its supports and the surrounding concrete is shown in P&ID 11405-A-5 through 11405-A-8, Figure 1.2-6, P&ID 11405-A-13 and 11405-A-14 inclusive.

During operation, the four pumps circulate water through the reactor vessel where it serves as both coolant and moderator for the core. The heated water enters the two steam generators, transferring heat to the secondary (steam) system, and then returns to the pumps to repeat the cycle.

System pressure is maintained by regulating the water temperature in the pressurizer where steam and water are held in thermal equilibrium. Steam is either formed by the pressurizer heaters or condensed by the pressurizer spray to limit the pressure variations caused by contraction or expansion of the reactor coolant.

Overpressure protection is provided by two power-operated relief valves and two ASME Code spring-loaded safety valves connected to the pressurizer. Steam discharged from the valves is condensed and cooled by water in a quench tank. In the unlikely event that the discharge exceeds the capacity of the quench tank, the tank is relieved to the containment atmosphere via the quench tank rupture disc. The quench tank is located at a level lower than the pressurizer. This ensures that any power-operated relief valve or pressurizer safety valve leakage from the pressurizer, or any discharge from these valves, drains to the quench tank.

The reactor coolant system and its associated controls were designed to accommodate plant step load changes of ± 10 percent of full power and ramp changes of ± 10 percent of full power per minute without reactor trip. The system will accept, without damage, a complete loss of load with reactor trip.

Reactor coolant leaves the containment building in controlled quantities for treatment in the chemical and volume control system (CVCS). Water which is removed from the CVCS is processed by the radioactive waste disposal system.

4.3.2 Interfaces With Other Systems

To maintain the reactor coolant system water chemistry within the limits described in Section 4.3.13, a feed and bleed operation is maintained by the CVCS during normal operation. Three nozzles, one outlet and two inlet, are provided on the reactor coolant piping for this operation.

An inlet nozzle is provided on each of four reactor vessel inlet (cold leg) pipes to allow injection of borated water into the reactor vessel by the safety injection system. An outlet nozzle is provided on the reactor vessel outlet (hot leg) pipe on loop 2. During plant cooldown, water is removed from the reactor coolant system via this nozzle, circulated through the shutdown cooling heat exchangers where it is cooled and then returned to the reactor coolant system through the safety injection inlet nozzles.

Drains from the reactor coolant piping to the radioactive waste disposal system are provided for draining the reactor coolant system for maintenance. A connection is also provided on the quench tank for draining it to the radioactive waste disposal system following a relief-valve or safety-valve discharge.

A Reactor Coolant Gas Vent System (RCGVS) is provided to vent the non-condensible gases from the Reactor Coolant Gas System (RCS). The non-condensible gases can be vented from the Reactor Vessel head or Pressurizer to the Quench Tank or the containment atmosphere during post-accident situations.

Sampling system lines are provided from the reactor coolant piping, the pressurizer surge line and the quench tank to the sampling room to provide a means for taking periodic samples of the coolant, pressurizer steam or quench tank contents for chemical and radiochemical analysis (see Section 9.13).

The Pressurizer Quench Tank is connected to the Nitrogen Supply System and is normally isolated. When necessary, the Nitrogen Blanket is maintained by manual operation of the Nitrogen Supply Valves.

A connection to the quench tank spray header from the demineralized water supply is provided for adding water to the quench tank. This water cools the tank following a pressurizer relief or safety discharge. It also restores the tank operating level after draining.

Component cooling water is supplied to the reactor coolant pumps. Part of the water is circulated through oil coolers to cool the bearing lubricating oil system. The remainder of the water flows through the thermal barrier and the pump integral heat exchanger, where it serves to keep the controlled bleed-off flow at approximately 130°F.

4.3.3 Reactor Vessel

The reactor vessel and top head assembly are shown in Figure 4.3-1. The reactor vessel and top head were designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class A. The requirements imposed on the reactor vessel design include those in Section III of the ASME code and those discussed in Section 4.3.3, 4.5.4, 4.5.5, and Appendix F of the USAR. The design parameters are listed in Table 4.3-1. The inner surface of the reactor vessel, which is in contact with reactor coolant, is clad with austenitic stainless steel. In the areas of internal attachments, the interior is clad with Ni-Cr-Fe alloy. The vessel closure flange is a forged ring with a machined ledge on the inside surface to support the reactor internals and the core. The flange was drilled and tapped to receive forty-eight 6.03 inch diameter closure stud bolts and was machined to provide a mating surface for the reactor vessel seal. A tapered transition section connects the flange to the cylindrical shell.

Table 4.3-1 - "Reactor Vessel Parameters"

Design Pressure, psia	2,500
Design Temperature, °F	650
Nozzles	
Inlet (4 ea), ID, in.	24
Outlet (2 ea), ID, in.	32
CEDM (41 ea), ID, in.	2.76
Instrumentation (6 ea) ID, in.	5.189
Head vent (1 ea), ID, in.	1.06
Dimensions	
Inside Diameter, minimum in.	140
Overall Height, Including CEDM Nozzles, in.	478-1/32
Height, Vessel Without Head, in.	385-7/8
Outside Diameter, in.	154-11/16
Wall Thickness, in.	7-11/32
Upper Head Thickness, in.	6-1/8
Lower Head Thickness, in.	3-11/16
Cladding Thickness, nominal, in.	7/32
Materials	
Shell	A-533, Grade B, Class 1 Steel
Forgings	A-508-64 Class 2
Cladding	Weld Deposited Type 304 SS
CEDM Nozzles	Ni-Cr-Fe Alloy
Instrumentation Nozzles	Ni-Cr-Fe Alloy
Dry Weights	
Head, lb.	91,586
Vessel, lb.	444,634
Studs, Nuts, & Washers, lb.	21,806
Flow Skirt, lb.	3,503
Total, lb.	561,529

Extra thickness in the vessel nozzle shell course provides required reinforcement for the nozzles. The nozzles are tapered internally to reduce coolant pressure losses. An internal boss around the outlet nozzles provides a mating surface for the core support barrel outlet nozzles. This boss and the outlet sleeve on the core support barrel are machined to a common contour to reduce core bypass leakage. A fixed hemispherical head is attached to the lower end of the shell. There are no penetrations in the lower head.

The removable upper closure head is hemispherical. The head flange is drilled to match the vessel flange closure stud bolt locations. The stud bolts are fitted with spherical washers located between the closure nuts and head flange to maintain stud alignment during flexing due to boltup. To ensure uniform loading of the closure seal, the studs are hydraulically tensioned with a special tool and checked with an elongation gage after tensioning.

Flange sealing is accomplished by a double-seal arrangement utilizing two silver-plated Ni-Cr-Fe alloy O-rings. The space between the two rings is monitored to allow detection of any inner ring leakage. The control element drive mechanism (CEDM) nozzles (Ni-Cr-Fe alloy through the head, stainless steel flanges) terminate with bolted and seal-welded flanges at the upper end which are aligned on a single plane. This arrangement standardizes control element assembly (CEA) extension shaft lengths and provides complete interchange ability of components. There are six instrumentation nozzles of similar construction to the control element drive mechanism nozzles. In addition to these nozzles there is a 1.06 inch diameter vent connection.

The core is supported from the reactor vessel flange. The control element drive mechanisms are supported by the nozzles in the reactor vessel head. Separate restraints are provided to absorb horizontal forces on the CEDM's during seismic disturbances. The reactor vessel is supported on four pads welded to the underside of the coolant inlet nozzles. This arrangement permits radial thermal growth of the vessel while maintaining it centered and restrained from movement resulting from seismic forces.

The design of the reactor vessel and its internals is such that with the nuclear steam supply system (NSSS) operating at 1500 MWt and an 77% load factor, the integrated fast neutron flux (E>1.0 Mev) will be less than 2.4×10^{19} n/cm^2 during the 40-year vessel design life and less than 1.49×10^{19} n/cm^2 (Reference 4-13) at the location of the critical reactor vessel beltline weld. This result was based upon surveillance materials tests and an expected reduced vessel fluence rate provided by the new core load designs beginning with fuel Cycle 8.

The reactor vessel internals are constructed with wetted parts of Stellite, Ni-Cr-Fe, stainless steel, or zircaloy. The control element drive mechanism housings, which act as a reactor coolant boundary, are stainless steel.

The vessel closure contains 48 studs, 6.03 inches in diameter, with eight threads per inch. The stud material is ASTM A540, Grade B24, with a minimum yield strength of 130,000 psi. The tensile stress in each stud when elongated for operational conditions is approximately 36.4 ksi. Calculations show that 34 uniformly distributed studs can fail before the closure will separate at design pressure. However, 16 uniformly distributed broken studs or four adjacent broken studs will cause O-ring leakage. Failure of at least 16 studs is necessary before the closure would fail by "zippering" open.

All areas of gross and local structural discontinuities of the vessel were analyzed for transient conditions. The analyses were performed in accordance with Paragraph N-415 of Section III, ASME Boiler and Pressure Vessel Code, and considered the combined effects of all specified mechanical and thermal transient loading conditions as given in Section 4.2.2 of the USAR. The areas of the vessel having the highest cumulative fatigue usage factors are summarized on Figure 4.3-2.

A program to document the chemical composition of reactor vessel belt line weld materials was completed in 1985. As part of this program, a search of records at Combustion Engineering's Chattanooga Materials and Metallurgical Laboratory yielded chemical analyses for some of the vessel weld wire heats and linked others to weld seams in the closure head which were later sampled and analyzed for chemical composition. Additional documentation for these welds was provided by surveillance weld data from Salem 2 and D.C. Cook. The copper and nickel compositions for the reactor vessel belt line weld and plate materials are displayed in Table 4.3-1a.

Table 4.3-1a - "Reactor Vessel Belt Line Material Chemical Compositions (Reference 4-14)"

Plate/Weld Identification	ID Number	Cu (w/o)	Ni (w/o)
Intermediate Shell Longitudinal Weld Seam	2-410	0.17	0.17
Power Shell Longitudinal Weld Seam	3-410	0.22	0.75
Intermediate to Lower Shell Girth Weld Seam	9-410	0.23	0.75
Intermediate Shell Plate	D-4802	0.12	0.56
Lower Shell Plate	D-4812	0.12	0.60

4.3.4 Steam Generators

The Nuclear Steam Supply System utilizes two steam generators, (Figure 4.3-3) to transfer the heat generated in the reactor coolant system to the secondary system. The design parameters for the steam generators are given in Table 4.3-2.

Table 4.3-2 - "Steam Generator Parameters"

Number	2
Type	Vertical U-Tube
Number of Tubes	5005
Tube Outside Diameter, in.	0.750
Nozzles and Manways	
Primary Inlet Nozzle (1 ea), ID, in.	32
Primary Outlet Nozzle (2 ea), ID, in.	24
Steam Nozzle (1 ea), ID, in.	26
Feedwater Nozzle (1 ea), ID, in.	14-5/16
Instrument Taps (9 ea), ID, in.	0.957
Primary Manways (2 ea), ID, in.	16
Secondary Manways (2 ea), ID, in.	16
Secondary Handhole (2 ea), ID, in.	5-11/16
Secondary Drain and Blowdown (1 ea), ID, in.	1.939
Primary Side Design	
Design Pressure, psia	2500
Design Temperature, °F	650
Design Thermal Power (NSSS), MWt	1500
Coolant Flow Rate (each), Nominal Operating, lb/hr	41.3×10^6
Nominal Operating Pressure, psia	2100
Secondary Side Design	
Design Pressure, psia	1000
Design Temperature, °F	550
Nominal Operating Steam Pressure, Full Load, psia	815
Nominal Operating Steam Temperature, Full Load, °F	520
Steam Moisture Content, Maximum, percent	0.20
Nominal Operating Blowdown Flow (each), Maximum, lb/hr	30,000
Design Thermal Power (NSSS), Btu/hr	2.560×10^9
Steam Flow (each), lb/hr	3.305×10^6
Feedwater Temperature, °F	444
Dimensions	
Overall Height, in.	647-1/16
Upper Shell Outside Diameter, in.	187-1/2
Lower Shell Outside Diameter, in.	126
Dry Weight, lb.	558,705
Flooded Weight, lb.	894,977
Operating Weight, lb.	681,424

Each steam generator is a vertical U-tube heat exchanger and was designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class A. The steam generators operate with the reactor coolant in the tube side and the secondary fluid in the shell side.

Hot reactor coolant enters a steam generator through the inlet nozzle, flows through 3/4-inch OD U-tubes, and leaves through two outlet nozzles. A vertical divider plate separates the inlet and outlet plenums. The plenums are stainless steel clad, while the primary side of the tube sheet is Ni-Cr-Fe clad. The vertical U-tubes are Ni-Cr-Fe alloy. The tube-to-tube sheet joint is welded on the primary side.

Feedwater enters a steam generator through the feed ring, mixes with the recirculating water from the steam separators, and flows down the annulus between the tube bundle wrapper and the steam generator shell.

Upon exit at the bottom, the secondary water is directed upward over the vertical U-tubes. Heat transfer from the primary side converts a portion of the secondary water into steam.

After leaving the vertical U-tube heat transfer surface, the steam-water mixture enters the centrifugal type separators. These impart a centrifugal motion to the mixture and separate the water particles from the steam. The water exits from the perforated separator housing and mixes with the feedwater. Final drying of the steam is accomplished by passage of the steam through corrugated plate dryers. The moisture content of the outlet steam is limited to a maximum of 0.2 percent at design flow.

The power-operated steam dump valves and steam bypass valve preclude opening of the safety valves following turbine and reactor trip from full power. The steam dump and bypass system is described in Section 10.

The steam generator shells are constructed primarily of SA-302, Grade B low alloy steel. Manways and handholes are provided for easy access to the steam generator internals.

Overpressure protection for the shell side of the steam generators and the main steam line piping up to the inlet of the turbine stop valves is provided by ten safety valves. These valves are ASME Code spring-loaded, open-bonnet, safety valves that discharge to atmosphere. Five safety valves are mounted on each of the main steam lines upstream of the steam line isolation valves but outside the containment. The opening pressure of the valves is set in accordance with ASME Code allowances. The valves can pass a steam flow equivalent to an NSSS power level of 1500 MWt at the nominal set pressure. Analyses which support Sections 14.9 and 14.10 (i.e., Refs. 14.9-1 and 14.10-1) are based on a minimum of four-of-five operable main steam safety valves on each main steam header during power operation. Parameters for the secondary safety valves are given in Table 4.3-3.

Table 4.3-3 - "Secondary Safety Valve Parameters"

Design Pressure, psia	1,000
Design Temperature, °F	550
Fluid - Saturated Steam	
Capacity, Eight Valves (each), lb/hr	794,062
Two Valves (each), lb/hr	126,299
Total Capacity, lb/hr	6.605×10^6
Set Pressure	
Two Valves, One per Unit, psia	1,050
Two Valves, One per Unit, psia	1,040
Two Valves, One per Unit, psia	1,025
Two Valves, One per Unit, psia	1,015
Two Valves, One per Unit, psia	1,000
Body Material	A-105, Gr II
Trim Material	Stainless Steel

The steam generators are mounted vertically on trapeze-like support structures which allow horizontal motion parallel to the hot leg due to thermal expansion of the reactor coolant piping. Stops are provided to limit this motion in case of a coolant pipe rupture. The top of each unit is restrained from sudden lateral movement by energy absorbers mounted rigidly to the concrete shield.

In addition to the transients listed in Section 4.2.2 each steam generator was also designed for the following conditions such that no component is stressed beyond the allowable limit as described in the ASME Boiler and Pressure Vessel Code, Section III:

a. 4000 cycles (2,000 each direction) of transient pressure differentials of 85 psi across the primary head divider plate due to starting and stopping the reactor coolant pumps.

b. 10 cycles of secondary side hydrostatic testing at 1235 psig while the primary side is at 0 psig.

c. 200 cycles of secondary side leak testing at 985 psig while the primary side is at 0 psig.

d. 5,000 cycles of adding 1000 gpm of 70°F feedwater with the plant in hot standby condition.

e. 80 cycles of adding 300 gpm of 32°F feedwater with the plant in hot standby condition.

In addition to the normal design transients listed above, and those listed in Section 4.2.2, the following abnormal transients were also considered in arriving at a satisfactory usage factor as defined in Section III of the ASME Boiler and Pressure Vessel Code.

a. 8 cycles of adding a maximum of 300 gpm of 32°F feedwater, with the steam generator secondary side dry and at 600°F.

The unit is capable of withstanding these conditions for the prescribed numbers of cycles in addition to the prescribed operating conditions without exceeding the allowable cumulative usage factor as prescribed in ASME Code, Section III.

4.3.5 Reactor Coolant Pumps

The reactor coolant is circulated by four pumps (Figure 4.3-4) which are of the vertical shaft, single-suction, single stage centrifugal type. The suction nozzles are in the bottom vertical position. The pressure containing components were designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class A.

The pump impeller is keyed and locked to the shaft. A close clearance thermal barrier assembly is mounted above the hydrostatic bearing. The assembly retards heat flow from the pump to the seal cavity located above the thermal barrier. The assembly also tends to isolate the hot fluid in the pump from the cooler fluid above, and in the event of a seal failure, serves as an additional barrier to reduce leakage from the pump. Each pump is equipped with replaceable casing wear rings. A hydrostatic bearing is located in the fluid between the impeller and thermal barrier to provide shaft support. Additional shaft support is provided by bearings in the electric motor which is connected directly to the pump shaft via a rigid coupling.

The shaft seal assembly is located above the thermal barrier and consists of four face-type mechanical seals, three full pressure seals mounted in tandem and a fourth low-pressure backup vapor seal designed to withstand operating system pressure with the pump stopped. The performance of the shaft seal system is monitored by pressure and temperature sensing devices in the seal system P&ID D-23866-210-111. Additional pressure transmitters have been installed in the leak off line between gaskets to annunciate a gasket failure condition. A controlled bleed-off flow through the pump seals is maintained to cool the seals and to equalize the pressure drop across each seal. The controlled bleed-off is collected and processed by the chemical and volume control system. Any leakage past the vapor seal (the last mechanical seal) is collected in the radioactive waste disposal system so that the pump leakage to the containment atmosphere is virtually zero. The seals are cooled by circulating the controlled leakage through a heat exchanger mounted integrally within the pump cover assembly; no damage would result in the event of pump operation without cooling water for up to 5 minutes. To reduce plant downtime and personnel exposure to radiation during seal maintenance, the seal system is contained in a cartridge which can be removed and replaced as a unit. The face seals can be replaced without draining the pump casing. The seal detail is shown in Figure 4.3-5.

There are two types of reactor coolant pump motors used at FCS. The original motors were supplied by General Electric with the flywheel surrounded by a cylindrical steel shroud. The flywheel assembly consists of three carbon steel discs keyed to the shaft above the motor. The shroud was designed to completely contain the largest conceivable missile in the event of a flywheel failure of 120% of the motor speed.

ABB Motor Flywheel

During the 1996 refueling outage, RC-3B motor was replaced with a motor manufactured by ABB Industries. The replacement motor was designed, manufactured and tested per the guidance of RG 1.14, Rev. 1, Reactor Coolant Pump Flywheel Integrity. The flywheel is a single piece design made from a forged ASTM A508 4/5 steel and shrink fit to the shaft collar. The flywheel is designed to withstand the largest predicted LOCA over speed of the motor. The flywheel is conservatively designed and made with close controlled quality material such that a flywheel failure is sufficiently small, therefore, a steel shroud was not included in the flywheel design.

The purpose of using the RG was to procure and install the reactor coolant pump motor as a component equal to or better than the design and construction of the original GE Motor. Contract 1977 provides the new motor specifications including the flywheel, Section 10.12. The flywheel was dedicated by EGS and is documented in Report No. SAIC-TR-751.200-02.

GE Motor Flywheel

The pump motor is provided with a flywheel which reduces the rate of flow decay upon loss of pump power. The inertia of the pump motor and flywheel is 70,000 lb-ft^2. The flywheel is surrounded by a cylindrical steel shroud to confine it in the event of a flywheel failure. Flow coast-down characteristics are discussed in Section 14.6. The steel shroud was designed to completely contain the largest conceivable missile in the incredible event of a flywheel failure. Conservative calculations were performed to evaluate the design wherein it was assumed that the motor reached 120 percent overspeed to establish the shroud thickness. This shroud thickness was then increased by 12 ½ percent. Although a concrete structure surrounds the pump motor on three sides, it is not considered to be a part of the missile shield if the flywheel fails. A cross section of the pump motor including principle dimensions is shown in Figure 4.3-6A. The flywheel housing is an intermediate transition section bolted to the top of the stator frame and to the bottom of the upper bracket. This heavy fabricated steel ring serves as an integral part of the stator supporting structure as well as a shroud for the flywheel. The masses and materials (including specifications) of major components using the part numbers in Figure 4.3-6A for identification, are as follows:

Part No.	Name	Mass	Material	Specification
23	Flywheel Housing	5,950 lbs.	Steel	ASTM A-284
26	Flywheel	11,500 lbs.	Steel	ASTM A-515 ASTM A-299

The addition of the shroud around the flywheel in the motor of the reactor coolant pump necessitates major disassembly of the motor for in-service inspection of the flywheel.

The flywheel assembly which has an inertia of 50,800 lb-ft^2, consists of three discs keyed to the shaft above the motor, with no mechanical interconnection. The dimensions of the discs are:

Outside Diameter, in.	70.
Inside Diameter, in.	13.5
Thickness, in.	3.67
Weight, ea., lbs.	3842.6

Conservative design bases and stringent quality control measures have been taken to preclude failure of the flywheel. As a result of these measures, it is considered that failure of a pump flywheel is unlikely.

A point also to consider is the probable sequence of events if the motor should exceed its design overspeed. Calculations based on minimum guaranteed material characteristics indicate that the rotor lamination would yield and fail at a lower speed than the flywheel. As rotor laminations increase in diameter due to yielding, they would contact the inside diameter of the stator punchings. Contact with the stator punchings would produce a high level of friction resulting in substantial braking torque that would limit overspeed. Rotor bar separation would occur at 2900 rpm. The flywheel disk failure would occur at 3300 rpm.

The selection of material, machining and manufacturing operations, quality control, and the rigorous acceptance criteria established to ensure the integrity of the flywheel and to minimize operating stresses include the following:

a. At least ½ inch of stock was left on the radius for machining during the flame cutting of the bore;

b. There are no stress concentrations such as stencil or punch marks or drilled or tapped holes within 8 inches of the edge of the flywheel bore;

c. Each flywheel plate was ultrasonically inspected in accordance with ASTM A-435 on 9 inch grid lines;

d. After balancing, the flywheel and motor assembly was tested at no load speed. The maximum allowable vibration for acceptance of the assembly was 1.5 mils.

The following design features ensure that the requirements for structural soundness were met:

a. Division of the mass into three separate discs;

b. A keyway fillet radius not less than 1/8 inch thereby minimizing stress concentrations;

c. Fabrication of the discs using forged carbon steel plate having different tensile strengths. (See Table 4.3-4 given below).

Table 4.3-4 - "GE Reactor Coolant Pump Flywheel Material"

	Disc #1	Disc #2	Disc #3	
Heat Numbers	B9696	X4878	C4985	B7980
Disc Location	Center	Top	Bottom	Bottom
Material Identification (Note 1)	ASTM A-515 (PVQ-67) Modified		ASTM A-299 (PVQ-67) Modified	
Chemistry				
C	0.06	0.26	0.27	0.27
Mn	0.32	0.66	1.30	1.34
P	0.01	0.01	0.014	0.01
S	0.021	0.026	0.02	0.02
Cu	0.13	0.20	0.15	0.09
Si	0.05	0.28	0.23	0.24
Ni	0.10	0.15	0.18	0.07
Cr	0.06	0.12	0.13	0.08
Mo	0.03	0.05	0.04	0.02
Al	0.088	0.01	0.008	0.008

Table 4.3-4 - (Continued)

Physicals				
Tensile Strength, ksi	48	74.3	88	86.5
Yield Strength .2% offset, ksi	31.7	45	57.5	51
% Elongation in 2 in.	24	26	26	28
Grain Size	8	2-4	4-6	
Heat Treatment	Note 2	Note 2	Note 3	
Bend Test	Note 4	Note 4	Note 4	
UT Test	Note 5	Note 5	Note 5	
NDT (Note 6)	30-60	30-60	30-60	

NOTE 1: ASTM material identification is a nominal identification. Material chemistry was modified by General Electric via PVQ-67 to create the specific yield properties that were obtained.

NOTE 2: 1600°F ± 25°F for ½ hour per inch of thickness and air cooled.

NOTE 3: Heated at 1625 to 1675°F for 1 hour per inch of thickness and then water quenched to 40°F followed by tempering at 1180°F for 1 hour per inch of thickness and air cooled.

NOTE 4: 180°F bend test performed with the ratio of Di of bend to thickness equal to 2. Test performed at room temperature with no cracks.

NOTE 5: The reactor coolant pump flywheels were given a shop ultrasonic inspection on a 9 inch grid, prior to assembly on the shaft. After assembly on the shaft, another shop ultrasonic inspection was performed, which covered 100% of the flywheel volume. This inspection was done from the top surface of the top disc, the bottom surface of the bottom disc, and the circumferences of all three disc segments. Subsequent inspections are performed at times when the motor is disassembled for maintenance purposes. These inspections will consist of visual inspections of the upper surface of the top disc and the bottom surface of the bottom disc and ultrasonic inspections from the circumferences of all disc segments.

The longitudinal beam examination from the periphery of each wheel section showed no significant ultrasonic indications in the keyway areas and no significant ultrasonic vertical indications throughout the wheel area.

NOTE 6: NDT values conservatively estimated from evaluation of similar material data contained in References 4-5 and 4-6.

The resistance to rupture of the reactor coolant pump flywheels has been examined at 120% overspeed. Using fracture mechanics data furnished by the motor vendor, the critical crack length for the disc most susceptible to crack propagation was found to be 3 inches assuming the crack extended radially outward from the keyway and penetrated completely through the thickness of the disc. Using the crack growth prediction techniques described in Reference 4-4, it is concluded that over 185,000 complete cycles from zero to 120% overspeed would be required to cause a ½ in. long crack extending radially from the keyway to grow to critical size.

The flywheel studies discussed in the preceding paragraph were based on a K_{IC} value of 60 ksi-in$^{1/2}$ was derived as follows. Since most of the published data on K_{IC} values for mild steel have been obtained at temperatures at or below the nil ductility temperature, a curve representing a lower bound of the available test data versus the difference between the nil ductility temperature and the test temperature was prepared (see Figure 4.3-12). This curve was used in combination with an upper bound estimate of the nil ductility temperatures expected for the pump flywheels (NDT=60°F, Ref. Table 4.3-4) to obtain the K_{IC} value expected for the pump flywheels at various operating temperatures.

The K_{IC} value of 60 ksi used for crack growth calculations corresponds to a minimum flywheel temperature of 65°F. The flywheel temperatures expected for normal operation would be at least 80 to 100°F and the corresponding values of K_{IC} expected would be 68-75 ksi.

The stresses in the flywheel at 120% overspeed were computed by the methods and equations presented in Reference 4-2.

The distribution of the stress normal to the radius (hoop stress) of the wheel is shown in Figure 4.3-13. The maximum stress at the bore is 17,500 psi. The stress intensity factor for various assumed crack lengths can be computed by several techniques. The most conservative technique is to assume that the maximum tensile stress acts over the entire crack length. Another technique which was shown in Reference 4-3 to produce accurate results, is the Irwin Method. For this method, the effect of the stress opening the crack is integrated over the crack length. A comparison of these results is shown in Figure 4.3-14. The use of the average stress (the mean of values at both ends) over the crack length results in a curve between the most conservative and the Irwin method curve. Using the average stress curve as an adequately conservative solution, it can be concluded that the critical stress intensity factor (toughness), K_{IC} required to prevent failure of a crack initially extending 3 inches from the keyway is 60 ksi-in$^{1/2}$. Toughness of the flywheel materials is greater than 75 ksi-in$^{1/2}$ at normal operating temperature as discussed above. The differences between the necessary and actual toughness values and the degree of conservatism of the analysis indicate that the flywheels would not fail at 120% over-speed, even if a crack existed which extended three inches from the keyway.

The fatigue crack growth can be conservatively assessed by assuming that each startup involves a 0 psi to 17,500 psi stress change (0 to 120% normal operational speed). The crack growth rate is related to the stress intensity factor by the equation:

$$\frac{d_a}{d_N} = C_{\circ} \Delta K_I^n$$

where $C_{\circ} = 2.3 \times 10^{-19}$ and $n = 3$. (Typical values are given in Reference 4-4 for material similar to that used for the flywheel). The results of this calculation indicate that a crack originally extending 0.5 inches from the keyway would grow to be 3 inches from the keyway in 185,000 full stress cycles. This value is orders of magnitude higher than the number of cycles expected during operation of the Fort Calhoun plant.

ABB Motor and GE Motor

ABB Motor Flywheel

The pump motor is provided with a flywheel which reduces the rate of flow decay upon loss of pump power. The inertia of the pump motor and flywheel is 70,000 (-0%, +5%)lb-ft^2. Flow down characteristics are discussed in Section 14.6. Below is a summary of the flywheel strength analysis and for a more detailed analysis see ABB Calculation HTAM622595 (OPPD Calc. FC06608, Flywheel Strength Analysis - Operating, Seismic and Fracture Conditions).

Material Specification

The material properties of the steel used for the flywheel are as follows:

Steel ASTM A508 Class 4/5 (forged)	Symbol	Value	Unit	Value	Unit
Elastic modulus	E	210000	N/mm^2	30479	ksi
Shear modulus	G	80000	N/mm^2	11611	ksi
Poisson's ratio	v	0.30	--	0.30	--
Mass density	ρ	7.85E-06	kg/mm^3	0.284	lb/in^2
Yield strength (min) specified	$R_{\rho 0.2}$	585	N/mm^2	85	ksi
Yield strength (min) measured	$R_{\rho 0.2}$	735	N/mm^2	106	ksi
Ultimate tensile strength specified	R_m	725-895	N/mm^2	105-130	ksi
Ultimate tensile strength measured	R_m	863	N/mm^2	125	ksi
Critical stress intensity factor specified	K_{ic}	3470	$[N/mm^2]*mm^{1/2}$	100	$ksi*in^{1/2}$
Critical stress intensity factor measured	K_{ic}	7148	$[N/mm^2]*mm^{1/2}$	206	$ksi*in^{1/2}$

Seismic Loading	Horizontal	Vertical
Operating Basis Earthquake (OBE):	±2.0 g	±3.0 g
Design Basis Earthquake (DBE):	±3.0 g	±3.0 g

Non-Ductile and Ductile Analysis

Critical Fracture Speed	Critical Fracture Speed rpm	Predicted LOCA Overspeed rpm	Safety Margin
Non-Ductile Fracture	4700	3697	1.27
Ductile Fracture	3910	3697	1.05

The flywheel is accessible for 100 percent in-place volumetric ultrasonic inspection with a minimum of motor disassembly. See Figure 4.3-6B for ABB Motor Cross-Section.

For more detail on the flywheel critical crack growth prediction, stress intensification factor determination, non-destructive examinations, and others, see Contract 1977, ABB Calc. HTAM622595 (OPPD FC06608), Flywheel Strength Analysis - Operating, Seismic and Fracture Conditions, ABB Calc. IMV_MB_9611 (OPPD FC06605), Deformation of the Flywheel as Consequence of a LBLOCA Overspeed Accident, EGS Report No. SAIC-TR-751.200-02 and associated drawings. The predicted LOCA Overspeed is summarized in ABB CE Memorandum ST-95-0714, RCP Overspeed Transient Data, 12-26-95.

The pump motor assembly includes motor bearing oil coolers, seal chamber, controls and instruments. An oil collection system has been installed to receive and remove any leakage from the pump motor, external oil cooler, flanged or gasketed connections, oil level sight glasses, fill connection points and reservoirs. The system also contains and removes to storage, any pressurized or potential unpressurized oil leak from any crack in the lubricating oil system external to the pump. Each pump bay is periodically inspected for signs of lube oil leaks and potential fire hazard concerns. The bays are cleaned of normal RCP lube oil accumulation on a periodic basis. Cooling water is provided from the component cooling water system.

The NRC has issued an Exemption (Reference 4.7.30 and 4.7.31) from certain requirements of 10 CFR Part 50 regarding the reactor coolant pump lube oil collection system. The staff concluded for the ABB motor located in B reactor coolant pump bay an oil collection system is not needed for:

1) The unpressurized upper bearing cooling water penetrations located 3.15" above the normal oil level,

2) The unpressurized lower bearing component cooling water penetrations located 1" above the normal oil level,

3) The unpressurized vent line on the lower bearing resistance temperature detector (RTD) located 2.4" above the normal oil level,

4) The unpressurized upper bearing RTD located 10" above the normal oil level, and

5) The unpressurized lower bearing oil level transmitter line.

The NRC also concluded that an exemption is not needed for:

1) The motor cooling air vents of RCP RC-3B (ABB Motor),

2) The anti-rotation device air vents and the motor cooling air vents of the remaining RCPs, or

3) The lack of a flash arrestor for the RCP oil collection system vent.

A mechanism is provided on each pump to prevent reverse rotation. This mechanism is a free-wheeling clutch device with sprags located between concentric inner and outer races. The design parameters for the reactor coolant pumps are given in Table 4.3-5.

Reverse rotation of a reactor coolant pump is sensed by a reverse rotation switch. This switch actuates an alarm in the control room. Reverse rotation indicates failure of the mechanical antireverse rotation device.
P&ID D-23866-210-111 identifies and shows the location of the reverse rotation switch on each of the four reactor coolant system pumps. The reverse rotation switches are OCS-3112, OCS-3132, OCS-3152 and OCS-3172 for pumps RC-3A, RC-3B, RC-3C, and RC-3D respectively.

Table 4.3-5 - "Reactor Coolant Pump Parameters"

Number	4	
Type	Vertical, Centrifugal Limited Leakage	
Shaft Seals	Mechanical (4)	
Stationary Face	Carbon CCP-72	
Rotating Face Body	A-362, Gr CF8	
Rotating Face Ring	Titanium Carbide	
Design Pressure, psia	2,500	
Design Temperature, °F	650	
Nominal Operating Pressure, psia	2,100	
Design Flow, gpm	47,500	
Design Head, ft.	225	
Maximum Flow (One-pump operating), gpm	59,800	
Dry Weight of Pump and Motor Assembly, lb.	111,100	
Nominal Operating Weight of Pump & Motor Assembly, lb.	115,300	
Reactor Coolant Volume, ft^3	69	
Materials		
Impeller	A-351, Gr CF8	
Shaft	A-182, Tp 304	
Casing	A-351, Gr CF8M	
Casing Wear Ring	A-362, Gr CF8	
Hydrostatic Bearing		
Bearing	A-362, Gr CF8 Col. #6 Overlay	
Journal	A-362, Gr CF8 Col. #5 Overlay	
Piping Connections		
Cooling Water (4 ea) nominal, in.	3	
Controlled Bleed-off (1 ea) nominal, in.	3/4	
Seal Leakage (1 ea) nominal, in.	3/4	
Primary Pressure Taps (2 ea) nominal, in.	3/4	
Seal Vent Pressure, Tap (3 ea) nominal, in.	3/4	
Motor	GE	ABB
Voltage, volts	4,000	4,000
Frequency, Hz	60	60
Phases	3	3
Horsepower/Speed, Hot, hp/rpm	2660/1185	/
Horsepower/Speed, Cold, hp/rpm	3650/1185	3650/1193 /(Full Load)
Instrumentation		
Seal Temperature Detectors	2	
Pump Casing Pressure Taps	2	
Seal Pressure Detectors	3	
Controlled Bleed-off Flow Rate Detectors	1	

Table 4.3-5 (Continued)

Instrumentation (Continued)	
Motor Oil Level Detectors	2
Motor Bearing Temperature Detectors	4
Motor Stator Temperature Detectors	6 installed (one used)
Reverse Rotation Detector	1
Vibration Detection System	9
Oil Lift Pressure Detector	1
Low Pressure Lube Oil Pressure Detector	1
Lube Oil Filter Delta-p Detector	1
Anti-Reverse Device Pressure Detector	1 (GE Motors Only)
Anti-Reverse Device Temperature Detector	1 (GE Motors Only)
Anti-Reverse Device Lube Oil Detector	1 (ABB Motor Only)
Total Seal Assembly Leakage (Nominal and Standby Operation)	
Three Seals Operating, gpm	1.0
Two Seals Operating, gpm	1.23
One Seal Operating, gpm	1.73

The reactor coolant pump and motor are supported by three support lugs welded to the pump casing. The pump is hung on rods from the overhead structure and spring hangers are employed between the support rods and the overhead. Movement in the horizontal plane to compensate for pipe thermal growth and contraction is permitted. Vertical movement is not restrained.

The pump is constructed of high alloy cast stainless steel parts to minimize corrosion. The mechanical seals consist of a rotating titanium carbide ring riding over a hard carbon face. These materials are listed in Table 4.3-5. The design life of the seal arrangement is at least 2 years. Each seal is designed to accept the full operating system pressure, but normally operate at one-third system pressure.

The pump performance curve is shown in Figure 4.3-7.

The air-cooled, self-ventilated pump motor is sized for continuous operation at the flows resulting from four-pump operation with 0.75 specific gravity water. The motor service factor is sufficient to allow continuous operation with 1.0 specific gravity water. The motors are designed to start and accelerate to speed under full load when 80 percent or more of their normal voltage is applied. The motors are contained within standard drip-proof enclosures and are equipped with electrical insulation suitable for a zero to 100 percent humidity and a radioactive environment of 30 R/hr.

The analytical techniques employed in stress analysis of the pump casings to assure that the design of the main coolant pump casings satisfy the design rules of the Section III of the ASME Boiler and Pressure Vessel Code were based upon the displacement (stiffness) formulation for structural analysis and incorporate finite element methods. These techniques were confirmed for pressure loading by experimental results obtained from strain gage test of a similar full scale pump case. Temperature distributions used in a stress analysis were based upon three-dimensional finite difference solutions of the differential equation of heat conduction for steady state and transient conditions. Stress intensities were derived from the results obtained from the above techniques for each specified loading condition and evaluated in accordance with the design rules of Section III of the ASME Boiler and Pressure Vessel Code.

4.3.6 Reactor Coolant Piping

The reactor coolant piping consists of 32-inch ID hot leg pipes from the reactor vessel outlets to the steam generator inlets and 24-inch ID cold leg pipes between the steam generator outlets to the pump suction nozzles and between the pump discharges and the reactor vessel inlets. The other major piece of reactor coolant piping is the 10-inch, schedule 160 surge line pipe between the pressurizer and the hot leg in loop 1. Design parameters for the reactor coolant piping are given in the piping list Table 4.3-6.

The reactor coolant piping was sized to obtain a coolant velocity which would provide a reasonable balance between erosion, corrosion, pressure drop and system volume. The surge line is sized to limit the frictional pressure loss through it during the maximum insurge so that the pressure differential between the pressurizer and the heat transfer loops is no more than 5 percent of the system design pressure.

The hot and cold leg pipes have no individual supports. The hot and cold legs are supported by connections to the steam generator, reactor vessel and reactor coolant pumps.

The reactor coolant piping is 316 stainless steel. The 10-inch surge line is also Type 316 stainless steel.

Thermal sleeves are installed in the surge nozzle, charging nozzle and shutdown cooling inlet nozzle to reduce thermal shock effects from auxiliary systems. All nozzles on the reactor coolant piping are constructed of stainless steel.

The piping was shop fabricated and shop welded into subassemblies to the greatest extent practicable to minimize the amount of field welding. Fabrication of piping and subassemblies was performed by shop personnel experienced in making large heavy wall welds. Welding procedures and operations met the requirements of Section IX of the ASME Boiler and Pressure Vessel Code. All welds were 100 percent radiographed and liquid-penetrant tested to the acceptance criteria of Section III of the ASME Boiler and Pressure Vessel Code. All reactor coolant piping penetrations were attached in accordance with the requirements of the USAS B31.1. Field welds were made to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Cleanliness standards consistent with nuclear service were maintained during fabrication and erection.

All small piping connected to the reactor coolant system, such as instrument lines, is standard welded schedule stainless steel using the same specification limits as the major piping connections.

Table 4.3-6 - "Piping List"

Line No.	Description	Material	Schedule	Nominal Size (inches)	Design Pressure (psig)	Design Temp (°F)
1	Reactor Vessel to Steam Generator	Type 316 Stainless Steel	Special 3-1/4 inch wall pipe and 3-1/2-inch wall elbows	32 ID	2485	650
2	Steam Generator to Reactor Vessel	Type 316 Stainless Steel	Special 2-1/2-inch wall pipe and 2-5/8-inch wall elbows	24 ID	2485	650
3	Surge Line, Hot Leg to Pressurizer	Type 316 Stainless Steel	160	10	2485	700
4	Pressurizer Spray	Type 316 Stainless Steel	160	3 and 4	2485	650
5	Pressurizer Power Operated Relief Valve Inlet	Type 316 Stainless Steel	160	3	2485	700
6	Primary System Drain Lines	Type 316 Stainless Steel	160	2	2485	650
7	Pressurizer Relief Line to Quench Tank	Type 304 Stainless Steel	40	6 and 8	350	650
8	Reactor Coolant Gas Vent System	Type 316 Stainless Steel	160	1	2485	650

4.3.7 Pressurizer

The pressurizer maintains reactor coolant system operating pressure and compensates for changes in coolant volume during load changes. Table 4.3-7 gives design parameters for the pressurizer. The pressurizer is shown in Figure 4.3-8.

Table 4.3-7 - "Pressurizer Parameters"

Design Pressure, psia	2,500
Design Temperature, °F	700
Nominal Operating Pressure, psia	2,100
Nominal Operating Temperature, °F	642.8
Internal Free Volume, ft^3	900
Nominal Water Volume, Full Power, ft^3	500
Nominal Steam Volume, Full Power, ft^3	400
Installed Heater Capacity, kW	900
Spray Flow, Maximum, gpm (Note 1)	279
Spray Flow, Continuous, gpm (Note 2)	3.0
Nozzles	
Surge Line (1 ea) ID, in.	8.5
Safety Valve (2 ea) ID, in.	2-5/8
Relief Valve (1 ea) ID, in.	2-5/8
Spray (1 ea) ID, in.	3.529
Heaters (72 ea) ID, in.	0.903
Instruments, Level (8 ea) ID, in.	0.75
Temperature (2 ea) ID, in.	0.75
Materials	
Vessel	A-533, Gr B, Class 1
Cladding	AISI-304 SS and Ni-Cr-Fe Alloy
Dimensions	
Overall Height, in.	399-1/4
Outside Diameter, in.	99-1/4
Inside Diameter, in.	90
Cladding Thickness, in.	7/32
Dry Weight, Including Heaters, lb.	123,015
Nominal Operating Weight, lb.	145,565

NOTE 1: Maximum Spray flow is based on both pressurizer spray valves

NOTE 2: Continuous spray flow is based on a flow of 1.5 gpm for each mini spray valve.

Pressure is controlled by maintaining the saturation temperature corresponding to the desired system pressure. At full load conditions, slightly more than one-half the pressurizer volume is occupied by saturated water, and the remainder by saturated steam. A number of the pressurizer heaters are operated continuously to offset spray and heat losses, thereby maintaining the steam and water in thermal equilibrium at the saturation temperature corresponding to the desired system pressure.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. A reactor coolant ventilation system is available to allow the operator to vent the pressurizer steam space. Although designed for accident conditions, the system may also be used to aid in the pre- or post-refueling venting. During load changes, reactor coolant temperature is manually controlled to vary as a function of load in accordance with approved operating procedures. A reduction in load will result in a lower reactor coolant temperature. The resulting contraction of the coolant lowers the pressurizer water level causing the reactor system pressure to fall. This loss of pressure is partially offset by flashing of pressurizer water into steam. All pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting pressure decrease. Should the water level in the pressurizer drop sufficiently below its setpoint, charging pumps in the chemical and volume control system are automatically started to add coolant to the system and restore pressurizer level.

When steam demand is increased, reactor coolant temperature is raised in accordance with approved operating procedures. The expanding coolant enters the pressurizer (insurge), compressing the steam and raising system pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. Should the pressure increase be large enough, the pressurizer spray valves open, spraying coolant from the reactor coolant pump discharge (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase. The programmed pressurizer water level is a power-dependent function. A high level signal produced by an insurge causes the letdown control valves to open, releasing coolant to the chemical and volume control system and restoring the pressurizer to the prescribed level.

Small pressure and coolant volume variations are accommodated by the steam volume which absorbs flow into the pressurizer and by the water volume which allows flow out of the pressurizer. The total volume of the pressurizer is determined by consideration of the following factors:

a. Sufficient water volume is necessary to prevent draining the pressurizer as the result of a reactor trip or an excess load incident. In order to preclude the initiation of safety injection and of automatic injection of concentrated boric acid by the charging pumps, the pressurizer is designed so that the minimum pressure observed during such transients is above the setpoint of the safety injection actuation signal;

b. The heaters must not be uncovered by the outsurge following load increases; 10 percent step increase, and 10 percent per minute ramp increases;

c. The steam volume must be sufficient to yield acceptable pressure response to normal system volume changes during load change transients;

d. Excess water volume over the amount actually needed adds to energy release during the maximum hypothetical accident and adds to the required containment volume;

e. The steam volume should be sufficient to accept the reactor coolant insurge resulting from loss of load without the water level reaching the safety valve or power operated relief valve nozzles;

f. During load following transients, the total coolant volume change and associated charging and letdown flow rates should be kept as small as practical and be compatible with the capacities of the volume control tank, charging pumps, and letdown control valves in the chemical and volume control system.

g. The pressurizer steam space requirements are located in Table 4.3-11a, step b and c (Reference 4-23).

To account for these factors and to provide adequate margin at all power levels, the water level in the pressurizer is programmed as a function of average coolant temperature as shown in Figure 4.3-10. High or low water level error signals result in the actions shown in Figure 4.3-11 and described above.

The pressurizer heaters are sized to heat the pressurizer at approximately 45°F/hr, when it is full of water. They are single-unit, sheath-type immersion heaters which protrude vertically into the pressurizer through sleeves welded in the lower head. Each heater is internally restrained to prevent high amplitude vibrations and can be individually removed for maintenance during plant shutdown. Approximately 17 percent of the heaters are connected to proportional controllers which adjust the heat input as required to account for steady-state losses and to maintain the desired steam pressure in the pressurizer.

The remaining backup heaters are connected to on-off controllers. These heaters are turned on by a low pressurizer pressure signal or high level error signal. This latter feature is provided since load increases result in an insurge of relatively cold coolant into the pressurizer, decreasing the temperature of the water volume. The action of the chemical and volume control system in restoring the level results in a pressure undershoot below the desired operating pressure. To minimize the pressure undershoot, the backup heaters are energized earlier in the transient, contributing more heat to the water before the low pressure setting is reached. A low-low pressurizer level signal de-energizes all heaters to prevent heater burnout.

The pressurizer spray system consists of pipes leading from the discharge side of reactor coolant pumps in loops 1B and 2A to the pressurizer spray nozzle. An automatic spray control valve in each of the lines controls the amount of spray by varying its position as a function of pressurizer pressure; both of the spray control valves function in response to the signal from the controller. These components are sized to use the differential pressure between the pump discharge and the pressurizer to pass the amount of spray required to prevent the pressurizer steam pressure from opening the power-operated relief valves during normal load-following transients. Use of a line from each of the heat transfer loops provides spray capability with less than four pumps operating. A small continuous flow is maintained through the spray lines to keep the spray lines and the surge line warm, reducing thermal shock during plant transients. This flow also aids in keeping the chemistry and boric acid concentration of the pressurizer water equal to that of the coolant in the heat transfer loops. An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant cooldown after the reactor coolant pumps must be shutdown due to low system pressure. To ensure that no steam enters the horizontal spray piping at the pressurizer top, a loop seal was installed. Steam is prevented from entering, since under certain operating conditions, the potential existed for mixing of relative cold water with steam from mini spray valves. This would have produced thermal stresses which could have led to a pipe fatigue and eventual leakage.

In the event of an abnormal transient which causes a sustained increase in pressurizer pressure, at a rate exceeding the control capacity of the spray, a high pressure trip level would be reached. This signal trips the reactor and opens the two power-operated relief valves. The steam discharged by the relief valves is piped to the quench tank where it is condensed.

In accordance with Section III of the ASME Boiler and Pressure Vessel Code, the reactor coolant system is protected from overpressure by two spring-loaded safety valves. These valves incorporate a loop seal inlet arrangement to limit leakage. The discharge from the safety valves is also piped to the quench tank.

A Reactor Coolant Gas Vent System (RCGVS) is available for use by the operator to vent non-condensible gases from the pressurizer to quench tank or containment atmosphere during a post-accident situation. Post-accident venting could be undertaken if the non-condensible gases were known to be interfering with core cooling or reactor coolant system pressure control. The system can also be used to vent noncondensible gases while filling or draining the Reactor Coolant System.

The pressurizer is supported by a cylindrical skirt welded to the lower head. Since the pressurizer surge line has sufficient flexibility, no provisions were made for horizontal movement and the skirt is bolted solidly to the pressurizer support structure.

The pressurizer is constructed of A-533, Grade B, Class 1 steel plate. The interior surface of the cylindrical shell and upper head is clad with stainless steel. The lower head is clad with a Ni-Cr-Fe alloy to facilitate welding of the Ni-Cr-Fe alloy heater sleeves to the shell. Stainless steel or Ni-Cr-Fe alloy safe ends were provided on the pressurizer nozzles as required to facilitate field welds to the connecting piping.

4.3.8 Quench Tank

The quench tank was designed to collect and condense the discharge from the pressurizer during normal operation and to collect the non-condensible gas discharge from the reactor vessel head or the pressurizer during a post-accident situation. In either case, the quench tank prevents reactor coolant system discharges from being released to the containment atmosphere. Parameters for the pressurizer quench tank are given in Table 4.3-8. Fabrication was in accordance with the ASME Boiler and Pressure Vessel Code, Section III.

The steam discharged from the pressurizer is discharged underwater by a sparger to enhance condensation by uniform distribution. The normal tank water volume of 520 cubic feet is sufficient to condense the total steam mass released by the relief valves during a zero to 112 percent reactor power swing, without reactor coolant letdown or pressurizer spray. The water temperature rise in the quench tank is limited to 80°F, assuming a maximum initial water temperature of 120°F. The gas volume in the tank is sufficient to limit the maximum tank pressure after the above steam release to 50 psig. The rupture disc setpoint is 75 psig, assuming a maximum initial gas pressure of 3 psig. The quench tank also has pressure relief capacity through a safety relief valve which vents when tank pressure reaches 70 psig. The valve discharge line is routed to the floor drain near the tank. The valve use is intended to minimize the possibility of diaphragm rupture since the tank would lose its oxygen-free blanket if a rupture were experienced. The quench tank is equipped with a demineralized water spray system to condense steam in the tank atmosphere and cool the tank water after a steam discharge into it. A drain and spray procedure is used to cool the tank after a discharge.

The quench tank can condense the steam discharged during a loss-of-load incident as described in Section 14.9 without exceeding the rupture disc setpoint, assuming normal blowdown of the safety valves at the end of the incident. It is not designed to accept a continuous safety valve discharge. The rupture disc vents to the containment atmosphere.

Table 4.3-8 - "Quench Tank Parameters"

Design Pressure, psig	100
Design Temperature, °F	340
Nominal Operating Pressure, psig	3
Nominal Operating Temperature, °F	104
Internal Volume, ft^3	700
Nominal Water Volume, ft^3	520
Nominal Gas Volume, ft^3	180
Blanket Gas	Nitrogen
Nozzles	
Pressurizer Relief, (1 ea), in.	8
Demineralized water, (1 ea), in.	2
Rupture Disc, (1 ea), in.	16
Drain, (1 ea), in.	2
Temp. Instrument, (1 ea), in.	1
Level Instrument, (2 ea), in.	½
Vent, (1 ea), in.	3/4
Materials	
Vessel	SA-212 Gr B
Coating	Phenoline No. 372
Dimensions	
Overall Length, ft-in.	15-6
Outside Diameter, ft.	8
Dry Weight, lb.	11,000
Normal Operating Weight, lb.	53,000

The tank is constructed of carbon steel with a phenolic coating on the interior surfaces. The tank normally contains demineralized water under a nitrogen overpressure. The sparger, spray header, nozzles and rupture disc fittings are stainless steel.

4.3.9 Valves

4.3.9.1 Actuator-Operated Throttling and Stop Valves

Parameters for the actuator-operated throttling valves for pressurizer spray are given in Table 4.3-9. Actuator-operated stop valve (power-operated relief isolation) parameters are given in Table 4.3-10. The position of each valve on loss of actuating signal (failure position) is selected to ensure safe operation of the system and plant. System redundancy is considered when defining the failure position of any given valve. Valve position indication is provided at the main control panel where considered necessary to ensure safe operation of the plant.

Table 4.3-9 - "Actuator-Operated Throttling Valve Parameters"

Design Temperature, °F	650
Design Pressure, psia	2,500
Maximum Flow (Total for both Valves), gpm	279
Valve Control Program (for 2100 psia setpoint)	
Valve Full Open, psia	2,225
Valve Closed, psia	2,175
Stem Leak-Off	Yes

Table 4.3-10 - "Actuator-Operated Stop Valve Parameters"

Design Temperature, °F	700
Design Pressure, psia	2,500
Actuator	Electric Motor
Failure Position	As Is
USASI Class	1,500 lb.

4.3.9.2 Manually Operated Valves

Valves in this category have backseats to limit stem leakage when in the open position. Globe valves are generally installed with flow entering the valve under the seat since this arrangement reduces stem leakage during normal operation or when closed.

4.3.9.3 Check Valves

All check valves are of the totally enclosed type. Pressure losses through the valves are conservatively taken as the maximum for a swing-type check at the given flows.

4.3.9.4 Power Operated Relief Valves (PORV)

The two power-operated relief valves (PORV) are provided to limit the lifting frequency of the ASME Code safety valves by relieving pressurizer steam at 150 psi below the nominal safety valve set point. The PORVs are actuated by the high system pressure reactor trip signal. The PORVs are also used to prevent over-pressurization of the reactor coolant system during operation at low temperatures, an operation mode when the nil ductility transition temperature (NDTT) becomes a consideration for structural integrity of the primary coolant pressure boundary. Parameters for these valves are given in Table 4.3-11.

Table 4.3-11 - "Pressurizer Power-Operated Relief Valve Parameters"

Design Pressure, psia	2,500
Design Temperature, °F	700
Fluid	Sat Stm, 0.1 Wt - % Boric Acid
Number	2
Capacity, minimum, (each), lb/hr	99,000
Type	Solenoid Operated
Set Pressure, psia	2350

The capacity of the power-operated relief valves is sufficient to pass the maximum steam surge associated with a continuous control element assembly withdrawal incident starting from low power. Assuming that a reactor trip is effected on a high-pressure signal, the capacity of the power-operated relief valves is sufficient to ensure that the pressurizer safety valves do not open. The relief valve capacity is also large enough so that the safety valves should not open during a loss-of-load incident from full power. This assumes normal operation of the pressurizer spray system, and reactor trip on high pressure.

The two half-capacity PORVs are located in parallel pipes which are connected to the single pressurizer relief valve nozzle on the inlet side and to the relief line piping to the quench tank on the outlet side. Each PORV line includes a motor operated block valve that is located upstream of the relief valve and serves as backup to isolate the PORV line in the event that the relief valve sticks open. The block valves can be operated manually from the control room.

The solenoid operated PORV can be selected to be operated either manually or automatically. When required, the operation of the PORV is automatic. At high pressure, the valves open at a preselected pressure sensed in the reactor coolant system and remain open until the pressure drops to a value below the preselected pressure. For NDTT protection, the PORV opens in the event a preselected low-pressure setpoint that indicates the reactor temperatures are below the NDTT limit is reached. When this system is enabled, the requirements of Table 4.3-11a are applicable (Reference 4-23). If necessary, manual operation of the PORVs can be accomplished from the control room regardless of the reactor coolant system temperature or pressure. A monitoring system, with readout in the control room furnishes position indication for PORV and ASME code safety relief valves.

The PORVs, block valves, and the associated control and power equipment are classified safety-grade to achieve greater valve reliability and to minimize the number of challenges to the operation of the emergency core cooling components and systems. The design provides the operator with the capability to control the operation of the PORVs and associated block valves when off site power is not available. The power supply for each PORV and the associated block valve is arranged to provide redundancy for each set of valves in the event of loss of offsite power. PORV PCV-102-1 is powered from Diesel Generator D1. The associated block valve, HCV-151, is powered from Diesel Generator D2. Similarly, PORV-102-2 is powered from Diesel Generator D2 and block valve HCV-150 is powered from Diesel Generator D1. Providing that both block valves are open at the time that loss of off site power occurs, operator control of each PORV relief path is assured by means of remote positioning of either the PORV or the block valve by power supplied from either Diesel Generator power bus.

Table 4.3-11a "Low Temperature Overpressure Protection Requirements"

a) Water solid operation allowable in the 82°F to 200°F temperature range only.

b) Pressurizer steam space of at least 30% required for temperatures between 200°F and 385°F (Low Temperature Overpressure Protection enable temperature).

c) The following applies whenever the first RCP will be started if all RCP's are idled:

1) Pressurizer pressure shall be maintained below the "Maximum Pressure for First Start RCP" curve.

And Either

2a) A pressurizer steam space of at least 53% shall be maintained,

Or

2b) If RCS cold leg temperature is below 200°F, then steam generator secondary side temperature must be less that 30°F above the RCS cold leg temperature.

d) Up to 3 RCP's can be operated at temperatures ≥210°F.

e) Only 2 RCP's can be operated below 210°F.

f) The shutdown cooling system must be aligned to the RCS at temperatures below 130°F during heatup or RCS pressure must be maintained below 250 psia is shutdown cooling is temporarily disabled.

g) The shutdown cooling system must be aligned to the RCS at temperatures below 250°F during cooldown or RCS pressure must be maintained below 250 psia if shutdown cooling is temporarily disabled.

h) Three HPSI pumps may be operable at temperatures above 385°F.

i) No more than two HPSI pumps may be operable at temperatures between 385°F and 320°F.

j) No more than one HPSI pump may be operable at temperatures between 320°F and 270°F.

k) No HPSI pumps may be operable below 270°F unless the flow is restricted to below the three charging pump flowrate and all three charging pumps are not operable. Putting the HPSI pumps in "pull-out" is assumed to disable them.

4.3.10 Pressurizer Safety Valves

Two safety valves located on the pressurizer provide overpressure protection for the reactor coolant system. They are totally enclosed, backpressure-compensated, spring-loaded safety valves meeting ASME Code requirements. Parameters for these valves are given in Table 4.3-12.

Table 4.3-12- "Pressurizer Safety Valve Parameters"

Design Pressure, psia	2,500
Design Temperature, °F	700
Fluid	Sat Stm, 0.1 Wt - % Boric Acid
Capacity, minimum, (each), lb/hr	200,000
Set Pressure	
RC-141, psig	2,530
RC-142, psig	2,485
Type	Totally Enclosed, Bellows
Accumulation, maximum, % of setpoint	3
Back Pressure, Compensation	Yes

The safety valves will pass sufficient pressurizer steam to limit the reactor coolant system pressure to 110 percent (2750 psia) of design pressure following a complete loss of turbine generator load with a reactor trip initiated by either high pressurizer pressure or thermal margin/low pressure while operating at 1,500 MWt. The reactor is assumed to trip on a high reactor coolant system pressure signal (Section 14.9). To determine the maximum steam flow, the only other pressure relieving system assumed operational is the secondary safety valves. Conservative values for all system parameters, delay times, and core moderator coefficient were assumed. Overpressure protection of the reactor coolant system is provided considering the effects of reactor coolant pump head, flow pressure drops, and elevation heads. The pressurizer safety valves discharge through the relief line piping into the quench tank.

Dynamic loadings provided by the safety valve manufacturer were used to develop the design of stops or snubbers to absorb the dynamic loads when these valves operate.

4.3.11 Missile and Seismic Protection

4.3.11.1 Missiles

The main coolant loops and the steam and feedwater piping are protected from missiles generated within the containment building. Barriers are provided where the use of radiation shielding and/or support structures for missile shielding would not be feasible for this purpose.

4.3.11.2 Seismic

The NSSS is designed to withstand the load imposed by the maximum hypothetical accident plus the load imposed by the maximum hypothetical earthquake without loss of function required for reactor shutdown and emergency core cooling.

4.3.12 Materials Exposed to Coolant

The materials exposed to the reactor coolant have shown satisfactory performance in operating reactor plants. A listing of materials is given in Table 4.3-13.

Table 4.3-13 - "Materials Exposed to Coolant"

Reactor	
Vessel Cladding	Weld Deposited Type 304 SS
Vessel Internals	304 SS and Ni-Cr-Fe Alloy
Fuel Cladding	Zircaloy-4
Control Element Drive Mechanisms	
Housings	348 SS
Gears	17-4 8 (Haynes No. 23)
Bearings	Stellite Ball Bearings
Piping	Austenitic Stainless Steel Type 316
Steam Generator	
Bottom Head Cladding	Type 304 SS
Tube Sheet Cladding	Ni-Cr-Fe Alloy
Tubes	Ni-Cr-Fe Alloy
Pumps	
Casing	A-351, Gr. CF8M
Internals	A-351, Gr. CF8
Pressurizer	
Cladding - Lower Head	Ni-Cr-Fe Alloy
Cladding - Shell and Top Head	AISI 304 SS
Base Metal (Surrounding TE-108 Nozzle)	A-533, Gr. B, Cl. 1

4.3.13 Insulation

Piping and equipment are insulated with a mass-type material compatible with the temperature and functions involved.

A removable metal reflective-type thermal insulation is provided on the flange stud area of the reactor vessel closure head to permit access to the head studs for removal and reinstallation of the head. The same type of insulation is also provided on the reactor vessel.

The thickness of insulation is such that the exterior surface temperature is not higher than approximately 20°F above the maximum containment ambient (120°F). Supports for the insulation, consisting of carbon steel rings formed to fit the OD of the respective shells, and necessary attachment brackets are provided. The heads of the vessels (excluding the reactor) have internally tapped studs appropriately spaced for attaching the insulation. All insulation support attachments were attached prior to final stress relief.

All mass-type insulation material is calcium silicate which has a low soluble chloride content and contains sodium silicate in order to minimize the possibility of chloride-induced stress corrosion of stainless steel.

4.3.14 System Chemical Treatment

Control and variation of the reactor coolant chemistry is a function of the chemical and volume control system. Sampling lines are provided from the reactor coolant piping to provide a means for taking periodic samples of the coolant for chemical analysis. Water chemistry nominal values during power operation are shown in Table 4.3-14.

Table 4.3-14 - "Reactor Coolant Chemistry "

Specific Conductivity, Prior to Additives, micromhos/cm (maximum)	40
pH (77°F)	4.5 to 10.2
Hydrogen, cc (STP) per Kg (H_2O)	27 to 50
Halogens	
Chlorides, ppm (maximum)	0.15
Fluorides, ppm (maximum)	0.10
Dissolved Oxygen, ppm (maximum)	0.1
Boric Acid	
Nominal, ppm	0 to 2,500
Maximum, ppm	15,000
LiOH	LiOH Program (Ref. 4-16)

The pH control is effected by adjusting the concentration of lithium hydroxide and boric acid. The solids content is maintained below the design level by minimizing corrosion through careful selection of materials, chemistry control, and purification of the letdown stream of reactor coolant through filters and demineralizers. Hydrogen is maintained in the reactor coolant to chemically combine with oxygen. Hydrazine may be added during initial startup for oxygen scavenging.

All wetted surfaces in the reactor coolant system are compatible with the above water chemistry.

4.3.15 System Leak Detection Method

The leak detection methods are intended to sense the leakage to the containment from the reactor coolant and auxiliary systems. Means are provided to locate the leakage and corrective action is taken to ensure that total leakage is below levels consistent with safe operation of the plant. Positive indications, in the control room, of reactor coolant leakage are provided by the air particulate monitor, the gas monitor, the specific humidity monitor and the sump level indicator monitor.

a. Containment Air Particulate Monitor RM-050

RM-050 takes continuous air samples from the containment atmosphere. The particulate activity is collected on a moving filter paper system. After passing through a noble gas monitor downstream of the particulate monitor, the sample is returned to the containment.

Further discussion on RM-050 is provided in Section 11.2.3.

b. Containment Gas Monitor RM-051

RM-051 indicates the presence of containment noble gas activity. It measures the gaseous radioactivity by continuously sampling the containment atmosphere.

Further discussion on RM-051 is provided in Section 11.2.3.

c. Dew Point Monitor

The specific humidity in the containment is related to the leakage from various equipment/systems. By determining the specific humidity, it is possible to deduce the amount of leakage originating inside the containment. In the absence of leaks which add water vapor to the atmosphere, the dew point will stabilize at the lowest temperature in the ventilation system cooling coil. In the absence of cooling water temperature change, an increase in dew point temperature indicates a leak inside the containment. This instrumentation is sensitive to an increase of leakage to the containment atmosphere on the order of 3 gpm per 1.5°F in dew point. This sensitivity is based on the assumption that the effluent from the leak mixes well within the containment atmosphere before being removed.

d. Containment Sump Pumps Operation

Sump pumps WD-3A & 3B each have a capacity of 50 gallons per minute. A system of floor drains connects all parts of the containment to the sump. Condensate collected on ventilation system cooling coils is fed directly into the containment sump. The containment sump level instruments, LT-599 and LT-600, provide a means to detect reactor coolant and auxiliary system leakage of approximately 1 gpm in 4 hours. Technical Data Book TDB-VII graph converts actual containment sump volume vs actual instrument level.

4.3.16 Primary to Secondary Leak Detection

Radiation monitors are provided to detect a primary to secondary leak. The condenser off-gas passes through radiation monitor RM-057 to serve as a steam generator leak detector. In addition, two in-line radiation monitors (RM-054A and RM-054B) are installed in the steam generator blowdown sample lines. These blowdown monitors would alarm if a primary to secondary leak occurred.

Further discussion on RM-054A, RM-054B and RM-057 is provided in Section 11.2.3.

4.5 TESTS AND INSPECTIONS

4.5.1 General

Shop inspection and tests of all major components was performed at the vendor's plant prior to shipment. An inspection at the site was performed to assure that no damage had occurred in transit. Testing of the reactor coolant system was performed at the site upon completion of plant construction. These tests included hydrostatic tests of all fluid systems. A complete visual inspection of all welds and joints was performed prior to the installation of the insulation. All field welds were radiographically and dye-penetrant inspected in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

A hot flow test was made of the reactor coolant loop up to zero power operating pressure and temperature without the core installed. The system was checked for vibration and cleanliness. Auxiliary systems were checked for performance (see Section 13).

4.5.2 Nil Ductility Transition Temperature Determination

The reactor vessel was designed and fabricated in such a manner that significant operational limitations would not be imposed on the reactor coolant system resulting from shifts in reactor vessel NDT temperature. The vessel material monitoring program was and is conducted within the guidelines of ASTM E-185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors." The preirradiated NDT temperature of the base plate material was established using drop weight tests in accordance with ASTM E-208 and correlations were made with Charpy impact specimen tests conducted in accordance with ASTM E-23. This correlation, along with the Charpy impact specimens irradiated in the surveillance program, were used to monitor vessel material NDT temperatures. For the drop weight tests, performed on unirradiated materials, the test temperature was selected to bracket the NDT temperatures of the material.

For the pre-irradiated Charpy tests, a minimum of three specimens of each material were tested at any one temperature. Tests were performed at a sufficient number of different temperatures to establish the energy-temperature curve.

The test material used in establishing the unirradiated NDT temperature of the base metal was obtained from (1/4) T (where T is plate thickness) and/or (3/4) T locations of sections of the plate used in the intermediate and lower shell courses. The thermal history of the plate from which the specimens were taken is representative of that of the shell plating. The impact properties at these locations were considered to be representative of the material through the plate. Since the NDT temperature of the material of the plate surface is lower than at (1/4) T, it was conservative to use the properties of (1/4) T to establish the initial minimum operating temperature and as the base for the minimum operating temperature after irradiation.

Table 4.5-1 is a NDT Data Summary of the results of the Charpy V-notch tests at 30-foot-lbs. and the drop weight tests for the reactor vessel plates and forgings in order to permit identification of the critical nil-ductility transition temperature for the vessel.

The C-E design curve for predicting the increase in NDTT as a function of neutron fluence was based on all available A 533 Grade B irradiation data and did not take into consideration the benefits of smaller increases in NDTT resulting from small amounts of residual and impurity elements in the reactor vessel materials. Samples of all plates in the core region of the reactor vessel as well as samples of deposited weld metal were analyzed chemically to determine the residual and impurity element content for future use in connection with this facility.

The shift in RT_{NDT} resulting from neutron irradiation damage to the reactor vessel beltline weld metals was recalculated using the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Damage to Reactor Vessel Material." (Reference 4-12)

Table 4.5-1 - "Fort Calhoun Reactor Vessel NDT Data Summary"

VESSEL LOCATION & CODE		DROP WEIGHT	CHARPY 'V' @ 30'#
FLANGE		0 °F	<-50 °F
NOZZLES			
INLET	1	-30 °F	<-100 °F
	2	-30 °F	<-40 °F
	3	-40 °F	<-90 °F
	4	-20 °F	<-4 °F
OUTLET			
	1	-30 °F	<-100 °F
	2	0 °F	+10 °F
UPPER SHELL			
	1	-20 °F	-26 °F
	2	-30 °F	-18 °F
	3	+10 °F	-3 °F
INTERMEDIATE SHELL			
	1	-50 °F	-33 °F
	2	-20 °F	-2 °F
	3	-30 °F	-2 °F
LOWER SHELL			
	1	-30 °F	-20 °F
	2	-20 °F	-22 °F
	3	-30 °F	+3 °F
BOTTOM HEAD			
PEEL		-60 °F	-45 °F
DOME		-50 °F	-30 °F
CLOSURE HEAD			
FLANGE		<-10 °F	<-50 °F
PEEL		-40 °F	+3 °F
DOME		-20 °F	+4 °F

4.5.3 Surveillance Program

The surveillance program was implemented to monitor the radiation-induced changes in the mechanical and impact properties of the pressure vessel materials. Changes in the impact properties of the material were evaluated by the comparison of pre- and post-irradiation Charpy impact test specimens. Changes in mechanical properties were evaluated by the comparison of pre- and post-irradiation data from tensile test specimens.

Three metallurgically different materials representative of the pressure vessel were investigated. These are base metal, weld metal and weld heat-affected zone (HAZ) material. In addition to the materials from the reactor vessel, material from a standard heat of A-533 which has been made available through the courtesy of the Heavy Section Steel Technology (HSST) Program was also used. This reference material was fully processed and heat treated and was used for Charpy impact specimens so that a comparison could be made between the irradiations in various operating power reactors and in experimental reactors. A complete record of the chemical analysis, fabrication history and mechanical properties of all surveillance test materials was maintained.

The exposure locations and a summary of the specimens at each location is presented in Table 4.5-2. The pre-irradiation NDTT of each plate in the intermediate and lower shell courses were determined from the drop weight tests.

Base metal test specimens were fabricated from sections of the shell plate in either the intermediate or the lower shell course which exhibits the highest unirradiated NDTT. All material for base test specimens were cut from the same shell plate.

The material used for the base metal test specimens were adjacent to the test material used for ASME Code Section III tests and were at least one plate thickness away from any quenched edge. This material was heat treated to a condition which is representative of the final heat treated condition of the base metal in the completed reactor vessel.

Weld metal and HAZ material were produced by welding together two plate sections from the intermediate or lower shell course of the reactor vessel. All HAZ test material was also fabricated from the plate which exhibits the highest unirradiated NDTT.

The material used for weld metal and HAZ test specimens was adjacent to the test material used for ASME Code Section III tests and was at least one plate thickness from any water-quenched edge. The procedures used for making the shell girth welds in the reactor vessel were followed in the preparation of the weld metal and HAZ test materials. The procedures for inspection of the reactor vessel welds were followed for inspection of the welds in the test materials. The welded plate was heat treated to a condition which is representative of the final heat treated condition of the completed reactor vessel.

Table 4.5-2 - "Summary of Specimens Provided for Each Exposure Location "

Capsule Location (a)	Base Metal Impact		Base Metal Tensile	Weld Metal Impact	Weld Metal Tensile	HAZ Impact	HAZ Tensile	Reference Impact (b)	Total Specimens Impact	Total Specimens Tensile
	L(c)	T(d)								
Vessel - 45°	12	12	3	12	3	12	3	-	48	9
Vessel - 85°	12	-	3	12	3	12	3	12	48	9
Vessel - 95°	12	6	3	12	3	12	3	6	48	9
Vessel - 225°	12	-	3	12	3	12	3	12	48	9
Vessel - 265°	12	12	3	12	3	12	3	-	48	9
Vessel - 275°	12	6	3	12	3	12	3	6	48	9
Totals	72	36	18	72	18	72	18	36	288	54

(a) Vessel specimens - located between thermal shield and reactor vessel

(b) Reference material correlation monitors

(c) L = longitudinal

(d) T = transverse

The test specimens are contained in irradiation capsule assemblies. These assemblies are located at approximately equally spaced radial positions near the reactor vessel. The axial position of the capsules will be bisected by the midplane of the core. The circumferential locations include the peak flux regions. The design of the surveillance capsule incorporates features which minimize the temperature differentials between the test specimens and the reactor environment. The capsule size and shape was chosen to minimize neutron flux and thermal and hydraulic perturbations within the surveillance capsules. The capsule design makes provisions for inclusion of fission threshold detectors and temperature monitors.

The location of the surveillance capsule assemblies is shown in Figure 4.5-1. A typical surveillance capsule assembly is shown in Figure 4.5-2. A typical Charpy impact compartment assembly is shown in Figure 4.5-3. A typical tensile monitor compartment assembly is shown in Figure 4.5-4.

Sufficient "archive" reactor vessel material was retained for the preparation of test specimens for at least two additional material surveillance capsules, that may be needed for additional monitoring in the event that thermal annealing becomes necessary to recover fracture toughness in the later years of vessel service. In addition to irradiation capsules and specimens for establishing pre-irradiation base line properties, archive samples of base metal, weld metal and weld heat affected zone materials were provided. The size of these samples is large enough to permit manufacture of a sufficient quantity of test specimens to fill at least two additional capsule assemblies.

Fission threshold detectors (U-238) were inserted into each surveillance capsule to measure the fast neutron flux. Threshold detectors of Ni, Ti, Fe, S, and CO-free Cu were also selected for this application to monitor the fast neutron spectrum.

The selection of threshold detectors was based on the recommendations of ASTM E-261, "Method for Measuring Neutron Flux by Radioactive Techniques." Activation of the specimen material is also analyzed to determine the amount of exposure.

The maximum temperature of the encapsulated specimens is monitored by including in the surveillance capsules small pieces of low-melting-point eutectic alloys or pure metals individually sealed in quartz tubes. These temperatures monitors are detailed in Table 4.5-3.

Table 4.5-3 - "Composition and Melting Points of Temperature Monitor Materials"

Composition, (wt %)	Melting Temperature, (°F)
80.0 Au, 20.0 Sn	536
90.0 Pb, 5.0 Sn, 5.0 Ag	558
97.5 Pb, 2.5 Ag	580
97.5 Pb, 0.75 Sn, 1.75 Ag	590

The temperature monitors provide an indication of the highest temperature to which the surveillance specimens were exposed but not the time-temperature history or the variance between the time-temperature history of different specimens. These factors, however, will affect the accuracy of the estimated vessel material NDT temperature to only a small extent.

The surveillance capsule assemblies are located between the thermal shield and the pressure vessel wall. These specimens receive, at any time, a slightly higher neutron dose than the pressure vessel. The NDT temperature shifts resulting from the irradiation of these specimens closely approximates the NDT temperature shift of the vessel materials.

Test specimens removed from the surveillance capsules are tested in accordance with ASTM Standard Test Methods for Tension and Impact Testing. The data obtained from testing the irradiated specimens is compared with the unirradiated data and an assessment of the neutron embrittlement of the pressure vessel material is then made. This assessment of the NDT temperature shift is based on the temperature shift in the average Charpy curves, the average curves being considered representative of the material.

The periodic analysis of the surveillance samples permits the monitoring of the neutron radiation effects upon the vessel materials. If, with due allowance for uncertainties in NDT temperature determination, the measured NDT temperature shift turns out to be greater than predicted, then appropriate limitations would be imposed on permissible operating pressure-temperature combinations and transients to ensure that the existing reactor vessel stresses are low enough to preclude brittle fracture failure.

The integrated fast neutron dose (fluence) of 2.4×10^{19} n/cm^2 (Reference 4-13) to the reactor vessel/clad interface at the end of life has been calculated using the methods described in Section 3.4.6. The value of 2.4×10^{19} n/cm^2 represents the maximum fluence to the reactor vessel/clad interface based upon the results of analysis of the 265° surveillance capsule material specimens and based upon a 40 year life with an 77% load factor at 1500 MWt full power rating. This surveillance capsule was removed for analysis at the end of Fuel Cycle 7. A more detailed description of the calculational method is described in "Evaluation of Irradiated Capsule W-265", TR-O-MCM-002, March, 1984 (Refer to section 3.4.6). The predicted change in NDTT as a function of vessel fluence is shown in Technical Specification Figure 2-3. An end of life neutron fluence of 1.49×10^{19} n/cm^2 (E<1 MeV) has been determined for the inside surface of the critical reactor vessel beltline weld. For further details refer to Section 2.1.2 of the Technical Specifications and Reference 4-12.

All surveillance capsules were inserted into their designated holders during the final reactor assembly operation. Each capsule will remain in the reactor until the desired fluence has been attained by the specimens within. Table 4.5-4 shows the target fluence values for each of the capsules and a tentative schedule for removal. The variations in the target fluence levels are due to the variations in the core power considering the three possible part length CEA positions. Replacement surveillance capsule assemblies were inserted into the 225° and 265° locations prior to fuel Cycle 8 and the new core configuration initiated at that time, to provide supplemental fluence information in the future.

Table 4.5-4 - "Capsule Removal Schedule (Ref. 4-13 and Ref. 4-17)"

Removal Sequence	Refueling Schedule (EFPY)	Capsule Removed	Capsule Target Fluence n/cm^2 (Measured Fluence) n/cm^2	Vessel Target Fluence n/cm^2 (Measured Fluence) n/cm^2
1	2.5	225°	(5.1×10^{18})	(3.4×10^{18})
2	5.9	265°	(9.0×10^{18})	(8.8×10^{18})
3	13.6	275°	(1.0×10^{19})	(1.0×10^{19})
4	20	45°	3.3×10^{19}	3.3×10^{19}
5	21	85°	2.0×10^{19}	2.0×10^{19}
6	27	95°	2.5×10^{19}	2.5×10^{19}
7	32	225°*	3.6×10^{19}	3.3×10^{19}
8	Standby	265°*	----	----
9	20	275°*	2.4×10^{18}	3.3×10^{19}

* Replacement assemblies were installed in the 225° and 265° locations following early removal of the 265° capsule. These capsules benchmark the change in core loading design initiated at 5.9 EFPY. Replacement capsule 275° was inserted at the end of Cycle 14 and contains the limiting weld material for the FCS vessel.

All refuelings are assumed to occur at 16 month intervals. The calculational uncertainty for all fluence values in Table 4.5-4 is ± 20 percent. Additionally, there can be a 30 percent variation in the fluence over the length of the capsule due to axial power variations.

4.5.4 Nondestructive Tests

Prior to and during fabrication of the reactor vessel, nondestructive tests based upon Section III of the ASME Boiler and Pressure Vessel Code were performed on all welds, forgings and plates as follows.

All full penetration pressure-containing welds were 100 percent radiographed to the standards of Section N-624.8 of Section III of the ASME Boiler and Pressure Vessel Code. Other pressure containing welds such as used for the attachment of mechanism housings, vents and instrument housings to the reactor vessel head were inspected by liquid-penetrant tests of the root passes and the final surface.

All forgings were inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings were tested using shear wave techniques. Rejection under longitudinal beam inspection, with calibration so that the first back reflection is at least 75 percent of screen height, was based on interpretation of indications causing complete loss of back reflection. Rejection under shear wave inspection was based on indications, exceeding in amplitude the indication from a calibration notch whose depth is 3 percent of the forging thickness, not exceeding 3/8-inch with a length of 1 inch.

All forgings were also subjected to magnetic-particle examination. Rejection was based on relevant indications of:

a. Any cracks and linear indications of an inclusion nature exceeding 3/4-inch;

b. Nonlinear indication with dimensions greater than 3/16-inch.

Plates were ultrasonically tested using longitudinal and shear wave ultrasonic testing techniques. Rejection under longitudinal beam testing performed in accordance with ASME Code Class 1338-2, with calibration so that the first back reflection is at least 50 percent of screen height, was based on defects causing complete loss of back reflection. Any defect which showed a total loss of back reflection which could not be contained within a circle whose diameter is the greater of 3 inches or one-half the plate thickness was unacceptable. Two or more defects smaller than described above which caused a complete loss of back reflection were unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect unless the defects were contained within the area described above. For shear wave testing, the maximum permissible flaw was one which did not exceed that from a calibrated notch having a depth of 3 percent of the plate thickness and 1-inch long.

Nondestructive testing of the vessel was performed during several stages of fabrication with strict quality control in critical areas such as constant calibration of test instruments, metallurgical inspection of all weld rod and wire, and strict adherence to the nondestructive testing requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The detection of flaws in irregular geometries was facilitated because most nondestructive testing of the materials was completed while the material was in its simplest form. Nondestructive inspection during fabrication was scheduled so that full penetration welds were capable of being radiographed to the extent required by Section III of the ASME Boiler and Pressure Vessel Code.

Each of the vessel studs received one ultrasonic test and one magnetic-particle inspection during the manufacturing process.

The ultrasonic test was a radial longitudinal beam inspection, and a discontinuity which caused an indication with a height which exceeded 20 percent of the height of the adjusted first back reflection was cause for rejection. Any discontinuity which prevented the production of a first back reflection of three quarters of the screen height was also cause for rejection.

The magnetic-particle inspection was performed on the finished studs. Axially aligned defects whose depths were greater than thread depth and nonaxial defects were unacceptable.

The vessel studs were stressed as they are elongated by the stud tensioners during the initial installation of the vessel head and are also at each refueling. The amount of elongation versus hydraulic pressure on the tensioner is compared with previous readings to detect any significant changes in the elongation properties of the studs. Studs which yield questionable data during the head installation, or receive damage to the threads, will be replaced before returning the vessel to pressure operations.

Table 4.5-5 summarizes the component inspection program during fabrication and construction.

Table 4.5-5 - "Reactor Coolant System Quality Assurance Program"

Reactor Vessel	
Forgings	
Flanges	UT, MT
Studs	UT, MT
Cladding	UT*, PT*
Nozzles	UT, MT
Plates	UT**, MT
Cladding	UT*, PT*
Welds	
Main Seams	RT, MT
CRD Head Nozzle Connection	PT
Instrumentation Nozzles	PT
Main Nozzles to Shell	RT, MT, UT*
Cladding	UT*, PT
Nozzle Safe Ends	RT, PT, UT*
Vessel Support Buildup	UT*, MT
All Welds - After Hydrostatic Test	MT
Steam Generator	
Tube Sheet	
Forging	UT, MT
Cladding	UT*, PT*
Primary Head	
Plate	UT**, MT
Cladding	UT*, PT*
Secondary Shell and Head	
Plates	UT**, MT
Tubes	UT, ET
Nozzles (Forgings)	UT, MT
Studs	UT, MT

* Above Code (Section III) (Winter 1965) (Winter 1967 Addenda for Reactor Vessel)

** A 100 percent volumetric ultrasonic test of plate for both two-directional shear wave and longitudinal wave was performed. The code requires that Section III, Class A plate receive only a longitudinal wave ultrasonic test on a 9 inch by 9 inch grid. The 100 percent volumetric ultrasonic test assures that plate used in the primary system is of the highest quality.

Table 4.5-5 - (Continued)

Steam Generator	
Welds	
Shell, Longitudinal	RT, MT
Shell, Circumferential	RT, MT
Cladding	UT*, PT
Nozzles to Shell	RT, MT
Tube-to-Tube Sheet	PT, GT*
Instrument Connections	MT, RT
Temporary Attachments After Removal	MT
All Welds - After Hydrostatic Test	MT
Nozzle Safe Ends	RT, (MT or PT)
Level Nozzles	MT
Pressurizer	
Heads	
Plates	UT**, MT
Cladding	UT*, PT*
Shell	
Plates	UT**, MT
Cladding	UT*, PT*
Heaters	
Tubing	UT, PT
Centering of Elements	RT*
Nozzle	UT, PT, MT
Studs	UT, MT
Welds	
Shell, Longitudinal	RT, MT
Shell Circumferential	RT, MT
Cladding	UT*, PT
Nozzles	RT, MT
Nozzle Safe Ends	RT, PT
Instrument Connections	PT
Support Skirt	RT, MT
Temporary Attachments After Removal	MT
All Welds After Hydrostatic Test	MT

* Above Code (Section III) (Winter 1965) (Winter 1967 Addenda for Reactor Vessel)

** A 100 percent volumetric ultrasonic test of plate for both two-directional shear wave and longitudinal wave was performed. The code requires that Section III, Class A plate receive only a longitudinal wave ultrasonic test on a 9 inch by 9 inch grid. The 100 percent volumetric ultrasonic test assures that plate used in a primary system is of the highest quality.

Table 4.5-5 (Continued)

<u>Pumps</u>	
Castings	RT, PT
Forgings	UT, PT
Welds	
Circumferential	RT, PT
Instrument Connections	PT
All Welds After Hydrostatic Test	PT
<u>Piping</u>	
Fittings (Static Casting)	RT, PT
Pipe (Casting)	RT, PT
Nozzles (Castings)	RT, PT
Welds	
Circumferential	RT, PT
Nozzle to Run Pipe	RT, PT
Instrument Connections	PT

RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current
GT - Gas Leak Test

4.5.5 Additional Tests

During design and fabrication of the reactor vessel, additional operations beyond the requirements of the ASME Boiler and Pressure Vessel Code Section III were performed by the vendor. Table 4.5-6 summarizes these additional tests.

During the design of the reactor vessel, detailed calculations were performed to ensure that the final product would have adequate design margins. A detailed fatigue analysis of the vessel for all design conditions was performed. In those areas which were not amenable to calculation, stress concentrations have been obtained through the use of photo-elastic models. In addition, Combustion Engineering performed test programs for the determination and verification of analytical solutions to thermal stress problems. Also, fracture mechanics and brittle fracture evaluations have been performed.

All materials used in the reactor vessel were carefully selected and precautions were taken by the vessel fabricator to ensure that all material specifications were adhered to. To assure compliance, the quality control staff of Combustion Engineering reviewed the mill test reports and the fabricator's testing procedures. Vendors of Class I equipment were limited to U.S.A. and Canadian companies. The Canadian companies involved were the Velan Valve Corporation of 2125 Ward Avenue, Montreal, Canada and Reuter-Stokes of 1315 Industrial Road, Preston, Ontario. Velan supplied primary system valves for Fort Calhoun and Reuter-Stokes supplied in-core detectors. Velan Valve Corporation manufactures high pressure valves and steam traps in America. The company has created suitable valves for such sophisticated applications as nuclear submarines and aircraft carriers including the U. S. S. Enterprise. Velan was authorized, at the time of construction at Fort Calhoun, by the ASME Boiler and Pressure Vessel Committee to use the "NPV" symbol of the American Society of Mechanical Engineers for nuclear line valves. Reuter-Stokes is a manufacturer of nuclear sensors and systems. They have supplied nuclear detectors to power stations in Europe, Asia and Canada . They also supplied the incore-flux monitoring system for the Consumer Power Company's Palisades Plant. Some stainless steel piping (2 inches and smaller) was procured from the Sandvik Steel Company, Sandvik, Sweden. Some of this piping was included in the reactor coolant boundary. Sandvik has supplied piping for several other U. S. nuclear power plants including Dresden, Peach Bottom, Palisades and Turkey Point. Full documentation in accordance with ANSI B31.7 was obtained from Sandvik.

Table 4.5-6 - "Reactor Coolant System Inspection, CE Requirements"

	CE Requirements	Code Requirement
Reactor Vessel		
Ultrasonic Testing (UT)	1. 100% Volumetric Longitudinal and Shear Wave UT of Plate Material	1. Longitudinal UT on a 9" x 9" Grid Pattern
	2. UT of Clad Bond to a 1 in^2 Unbonded Area Repair Standard	2. None
Dye-Penetrant Testing (PT)	1. PT Test Root each 1/2 in. and Final Layer of Welds for Partial Penetration Welds to Control Rod Drive Mechanism Head Adapters and Instrument Tube Connections	1. PT Test of Finished Weld
	2. PT Test Finished Surface of Cladding	2. None

Table 4.5-6 -(Continued)

	CE Requirements	Code Requirement
Reactor Vessel		
Ultrasonic Test (UT)	1. 100% Volumetric Longitudinal and Shear Wave UT of Plate Material	1. Longitudinal UT on a (UT) 9" x 9" Grid Pattern
	2. UT for Defects in Tubesheet Clad	2. None
Dye-Penetrant Testing (PT)	1. PT Test Finished Surface of Primary Head Cladding	1. None
Pressurizer		
Ultrasonic (UT)	1. 100% Volumetric Longitudinal and Shear Wave UT of Plate Material	1. Longitudinal UT on a Testing 9" x 9" Grid Pattern
Dye-Penetrant Testing (PT)	1. PT Test on Finished Surface face of Cladding	1. None
Radiography (RT)	1. Radiograph Heaters to Check Heater Wire Positioning	1. None

All welding methods, materials, techniques, and inspections complied with Sections III and IX of the ASME Boiler and Pressure Vessel Code. Before fabrication was begun, detailed qualified welding procedures, including methods of joint preparation, together with certified procedure qualification test reports, were prepared. Also, prior to fabrication, certified performance qualification tests were obtained for each welder and welding operator. Quality control was exercised for all welding rod and wire by subjection to a complete and thorough testing program in order to ensure maximum quality of welded joints. No Class I systems or components were fabricated using electroslag welding.

During the manufacture of the reactor vessel, in addition to the areas which were covered by the ASME Boiler and Pressure Vessel Code Section III, quality control by the vendor included:

a. Preparation of detailed purchase specifications which included cooling rates for test samples;

b. Vacuum degassing for all ferritic plates and forgings;

c. Specification of fabrication instructions for plates and forgings to provide control of material prior to receipt and during fabrication;

d. Use of written instructions and manufacturing procedures which enable continual review based on past and current manufacturing experiences;

e. Performance of chemical analysis of welding electrodes, welding wire, and materials for automatic welding, thereby providing continuous control over welding material;

f. The determination of NDT temperature through use of drop weight testing methods as well as Charpy impact tests;

g. Test programs on fabrication of plates up to 15 inches thick to provide information concerning material properties as thickness increases.

Shear wave and longitudinal wave ultrasonic testing was performed on 100 percent of all plate material.

Cladding for the reactor vessel is a continuous integral surface of corrosion-resistant material, 7/32-inch nominal thickness. The detailed procedure used, i.e., type of weld rod, welding position, speed of welding, nondestructive testing requirements, etc., was in compliance with the ASME Boiler and Pressure Vessel Code. One hundred percent ultrasonic testing of the reactor vessel cladding was performed.

Upon completion of all postweld heat treatments, the reactor vessel was hydrostatically tested, after which all weld surfaces, including those of welds used to repair material, were magnetic-particle inspected in accordance with Section III, Paragraph N-618 of the ASME Boiler and Pressure Vessel Code.

Surveillance of the quality control program was also carried out during the manufacture of the vessel by the Windsor Quality Control Section of Combustion Engineering and, for the applicant, by an independent consultant. This work included independent review of radiographs, magnetic-particle tests, ultrasonic tests and dye-penetrant tests conducted during the manufacture of the vessel. A review of material certifications, and vendor manufacturing and testing procedures was also conducted. Manufacturers' records such as heat-treat logs, personnel qualification files and deviation files were also included in this review.

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4.5.6 In-service Inspection

THE FOLLOWING TEXT (SECTION 4.5.6) IS AS THE IN-SERVICE INSPECTION PROGRAM WAS ORIGINALLY ENVISIONED. THE CURRENT INSPECTION PROGRAM IS CONTAINED IN THE FOLLOWING NRC-APPROVED PUBLICATIONS:

FORT CALHOUN STATION, UNIT NO. 1
10-YEAR IN-SERVICE EXAMINATION PLAN
FOR CLASS 1, 2, and 3 COMPONENTS

and

FORT CALHOUN STATION UNIT NO. 1
IN-SERVICE INSPECTION PROGRAM PLAN

4.5.6.1 Introduction

The proposed program is based on Section XI of the ASME Boiler and Pressure Vessel Code, In-service Inspection of Nuclear Reactor Coolant Systems. The Fort Calhoun nuclear steam supply system is a pressurized-water system supplied by Combustion Engineering. This plant was not specifically designed to meet the requirements of Section XI of the Code; therefore, 100 percent compliance may not be feasible or practical. However, access for in-service inspection was considered during the design and modifications have been made where practical to make provision for maximum access within the limits of the current plant design.

The program Table 4.5-7, shows the inspections to be performed within the first 5 years of plant operation. A tentative 10-year program has been planned, and if no difficulties are encountered, the tentative program will be continued. It should be emphasized that although there is a tentative program for a 10-year period, the initial commitment is for the first 5 years of the inspection program. After 5 years, the program will be reviewed in light of the technology and experience available at that time.

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At the present time, the most highly developed technique for volumetric inspection of reactor pressure vessel welds is ultrasonics. Both ultrasonic techniques and radiography can be used for volumetric inspection of most of the other components, but ultrasonic techniques will probably be used for the following reasons:

a. The pipe or vessel need not be drained.

b. The inspection can be mechanized, reducing radiation exposure to personnel.

c. Less time is usually required than for radiographic inspection.

d. Radiation levels and component geometry make radio graphy more difficult and sometimes impossible.

e. Adequate records can be obtained using manual techniques, and reproducible records can be obtained with mechanized techniques.

Omaha Public Power District has considered problems associated within-service inspection during design of the plant. These considerations have provided increased access such as at the main coolant nozzle-to-pipe welds. These considerations are also reflected in protection of sensitized wrought stainless steel safe ends from the reactor pressure vessel, steam generator, and pressurizer by the application of weld metal overlay on the ID and OD.

Although ultrasonic techniques will be used for most of the volumetric inspection, radiography may be used on piping and other areas where ultrasonic techniques cannot be used.

The method of inspection planned for each area -- volumetric, surface, or visual -- is shown in Table 4.5-7, Inspection Program. The detailed procedures are not shown but will be prepared at the time of the preoperational inspection order to permit incorporation of the latest available techniques.

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It should be noted that because of the plant design, there are conflicts with the exclusion criteria of the Code, and certain components and piping have been excluded.

4.5.6.2 Inspection Program

This program is planned for the first 5-year period, after which the program will be reviewed to take into account changes in technology, equipment, and the Code. This program complies with the intent of the Code but has variations due to plant design. One such variation is the control element drive mechanism, (CEDM) housing-to-head welds. These are partial penetration welds and the housing contains a thermal sleeve. Due to the thermal sleeve, ultrasonic inspection is not possible from the ID and geometry makes inspection impractical from the OD. Therefore, these penetrations will be visually inspected for leaks in lieu of a volumetric inspection.

Discussed below are the inspection programs for the reactor pressure vessel, the pressurizer, the steam generator, reactor coolant and other piping, pumps, and valves. Each system is discussed in categories corresponding to Section XI of the Code.

(1.0) Reactor Pressure Vessel

(1.1) Category A - Pressure Containing Welds in Reactor Vessel Belt-Line Region

It is intended that these welds be volumetrically inspected when required, using ultrasonic techniques. An inspection can be made from the ID by removing the vessel internals or from the OD by pre-placed equipment. These welds were inspected by multiple nondestructive testing techniques during manufacture and, in addition, will be subjected to a preservice volumetric inspection.

No inspection is planned during the first 5-year period; the inspection tentatively scheduled for the 10-year period is shown in Table 4.5-7.

(1.2) Category B - Pressure-Containing Welds in Remainder of Vessel

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The same technique used for inspection of the welds in the core region can be used to inspect the other circumferential welds in the barrel section of the reactor pressure vessel. Excluded from inspection in this category are the welds in the closure head that are within the control element drive shroud assembly. Efforts will be made to develop techniques to inspect these welds. If these efforts are successful, the excluded welds will be included in a revised program.

No inspection is scheduled to be performed in the first 5-year period; the inspection tentatively scheduled for the 10-year period is shown in Table 4.5-7.

(1.3) Category C - Pressure-Containing Welds, Vessel-to-Flange and Head-to-Flange

The head-to-flange weld can be inspected using either mechanized or manual ultrasonic techniques.

The inspection scheduled for the first 5-year period and tentatively scheduled for the 10-year period is shown in Table 4.5-7.

It is planned to inspect the vessel-to-flange weld using mechanized ultrasonic techniques. This inspection would also include the ligaments in the flange between the bolt holes.

The inspections scheduled to be performed in the first 5-year period and tentatively scheduled for the 10-year period are shown in Table 4.5-7.

(1.4) Category D - Pressure-Containing Nozzles in Vessels

These nozzles include 2 outlet and 4 inlet nozzles. The nozzle-to-shell weld of the outlet nozzles can be inspected from the ID without removal of the lower internal package or from the OD by preplaced tracks. A volumetric inspection of the integral extension of these nozzles inside the vessel can be made from the nozzle ID.

Inspection of the coolant inlet nozzles can be accomplished from the OD by pre-placed track; inspection from the ID requires removal of the vessel lower internals.

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The inspections scheduled to be performed in the first 5-year period and tentatively scheduled for the 10-year period are shown in Table 4.5-7.

(1.5) Category E-1 - Pressure-Containing Welds in Vessel Penetrations

The only penetrations in this category are the control element drive mechanism (CEDM) assemblies in the upper head. The CEDM assemblies have permanently attached thermal sleeves preventing volumetric inspection from the ID. There is no available technique for volumetric inspection from the OD; therefore, these penetrations are to be included in Category E-2.

(1.6) Category E-2 - Pressure-Containing Welds in Vessel Penetrations

A visual inspection for evidence of leaking will be made of the penetrations in the upper head at the time of the system hydrostatic test. The visual inspection for leakage can be made without removal of insulation.

The inspections scheduled to be performed in the first 5-year period and tentatively scheduled for the 10-year period are shown in Table 4.5-7.

(1.7) Category F - Pressure-Containing Dissimilar Metal Welds

There are dissimilar metal welds between the carbon steel main coolant nozzle forgings and the safe-ends. Ultrasonic inspection of the dissimilar metal welds on the outlet nozzles can be inspected from the ID after removal of the upper internal package or from the OD using preplaced track. The dissimilar metal welds on the inlet nozzles can be inspected from the ID when the vessel lower internals are removed or from the OD using preplaced track. Experience with other plants has shown that, in general, these welds can be volumetrically inspected with ultrasonic techniques. The feasibility of this inspection will be determined on the preoperational inspection, and this will determine whether an in-service inspection is feasible. Plant design does not provide access to the OD of these welds for currently available radiographic inspection.

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These dissimilar metal welds are planned to be inspected at the same time as the corresponding nozzle welds, as shown in Table 4.5-7.

(1.8, 1.9 & 1.10) Category G-1 - Pressure-Retaining Bolting

All of the reactor vessel studs are scheduled to be removed at each refueling cycle. They are, thus, available for a volumetric, surface, and visual examination as may be required.

The closure stud nuts can be examined with techniques similar to those for the studs, and the washers would be visually inspected only.

The ligaments between the bolt holes of the reactor pressure vessel would be volumetrically examined at the same time the vessel flange weld is examined.

The inspections scheduled to be performed in the first 5-year period and tentatively scheduled for the 10-year period are shown in Table 4.5-7.

(1.11) Category G-2 Pressure Retaining Bolting

The CEDM nozzle bolts will be inspected visually in place unless the CEDM flange is disassembled for another reason in which case the bolts will be visually inspected individually.

The inspections scheduled to be performed in the first 5-year period and tentatively scheduled for the 10-year period are shown in Table 4.5-7.

(1.12) Category H - Vessel External Supports

The reactor pressure vessel is supported by supports integrally welded to the nozzle forgings. The nozzle supports will be inspected with the weld in Category D as provided by the Code.

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(1.13 & 1.14) Category I - Vessel Interior Clad Surfaces

The Code requires that specified patches of cladding be prepared to facilitate inspection and be located in areas with identifiable indications insofar as practical. The requirements for inspection of the reactor vessel cladding are visual, but other techniques are being investigated and will be used if proven practical.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are shown in Table 4.5-7.

Selected areas of the cladding in the upper closure head will also be inspected preoperationally and in service, either visually and surface or using ultrasonic techniques from the OD of the head.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(1.15) Category N - Interior Surfaces and Internal Components of Reactor Vessels

It is proposed that inspections for this category be made by remote television or borescopic examination. A critical inspection will be made at the first refueling to detect if any changes have occurred due to initial operation. The amount of inspection to be performed at subsequent refueling outages will depend upon the results of the first inspection and those made on comparable pressurized-water systems.

The inspections to be performed during the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(2.0) Pressurizer

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(2.1) Category B - Pressure-Containing Welds in Vessels

The primary heads on the pressurizer are plate and nozzle-to-head welds and the circumferential weld joining the heads to the barrel section require inspection. There are also circumferential and longitudinal welds in the barrel section which require inspection.

The inspections scheduled for the 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(2.2) Category D - Pressure-Containing Nozzles in Vessels

The nozzle to shell weld and the inner nozzle radii of the nozzles in the upper head can be inspected from the OD of the pressurizer or from the ID of the nozzle. The nozzle to shell weld and the inner nozzle radii of the outlet nozzle in the lower head can be inspected from the head OD with difficulty due to the heater assemblies; the thermal liner in the nozzle prohibits inspection from the ID.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(2.3 & 2.4) Category E - Pressure-Containing Welds in Vessel

The instrument and sample penetrations and the heater connections of the pressurizer meet the exclusion criteria of Paragraph 120(d) of the ASME Section XI "Rules for In-service Inspection of Nuclear Reactor Coolant Systems".

The inspections scheduled for the 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(No Reference) Category F - Pressure-Containing Dissimilar Metal Welds

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There are dissimilar metal welds between the pressurizer and the safe-ends of the surge line in the lower head and the piping in the upper head. Experience with other plants has shown that, in general, these welds can be volumetrically inspected with ultrasonic techniques. The feasibility of this inspection will be determined on the preoperational inspection and this will determine whether the in-service inspection will be by ultrasonic or radiographic techniques.

Although the inspection of these items is not required by the Code, the inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(2.5 & 2.6) Category G - Pressure-Retaining Bolting

There is no bolting on the pressurizer 2 inches and above in diameter, but there is bolting less than 2 inches in diameter which requires visual inspection.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(2.7) Category H - Vessel External Supports

The pressurizer is supported by a skirt which is integrally welded to the pressurizer.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(2.8) Category I - Vessel Interior Clad Surface

A visual inspection of a 36-square-inch area of the clad surface of the pressurizer will be made at such time as the pressurizer is opened for other reasons.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7

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(3.0) <u>Steam Generators</u>

(3.1) <u>Category B - Pressure-Containing Welds in Vessels</u>

The primary heads of the steam generators are formed and welded plate, and there are welds to be inspected in the support stand-to-head and also the circumferential welds joining the heads to the tube sheet.

The inspections scheduled to be performed in the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(3.2) <u>Category D - Pressure-Containing Nozzles in Vessel</u>

The nozzles are integrally welded to the head. The inner nozzle radii of the heads can be inspected from the OD or from the ID of the steam generators, but the ID inspection is estimated to require an excessive exposure of personnel. A volumetric inspection of the nozzle-to-head welds and of the inner radii will be made from the OD. A visual inspection of the radii will be made at such time as the steam generators are opened for other reasons if radiation levels permit.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

<u>Category E-2 - Pressure-Containing Welds in Vessel Penetrations</u>

There are no pressure-containing penetrations in the lower head of the primary side.

(3.3) <u>Category F - Pressure-Containing Dissimilar Metal Welds</u>

There are dissimilar metal welds in the primary side of the steam generators joining the steam generator to the safe-ends. Experience with other plants has shown that, in general, these welds cannot be inspected with ultrasonic techniques. The method of in-service inspection will be determined on the preservice inspections.

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The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(3.4 & 3.5) Category G - Pressure-Retaining Bolting

The bolting on the steam generators is less than 2 inches in diameter.

There are no inspections scheduled in the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(3.6) Category H - Vessel External Supports

The steam generators are supported by lugs integrally welded to the lower head. These welds will be volumetrically inspected as required by the Code.

The inspections scheduled in the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(3.7) Category I - Vessel Interior Clad Surface

To open a steam generator to perform a visual inspection of the clad surface on the primary heads of the steam generators from the ID is estimated to require more exposure to personnel than is believed warranted by the results of such an inspection. However, a visual inspection will be made at such time as the generators are opened and decontaminated for other purposes.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(4.0) Piping Pressure Boundaries

(4.1) Category F - Vessel, Pump and Valve Safe-Ends to Primary Pipe Welds and Safe-Ends in Branch Piping Welds

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The reactor coolant piping of this plant is stainless steel and dissimilar metal welds occur in the reactor coolant pipe at attachments to carbon steel components. The majority of these welds have been covered in categories associated with the components. Experience with other plants has shown that, in general, these welds can be volumetrically inspected with ultrasonic techniques. The feasibility of this inspection will be determined by the preoperational inspection, and this will determine whether the in-service inspection will be by ultrasonic or by radiographic techniques.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(4.2) Category J - Pressure-Containing Welds in Piping

The circumferential welds in the reactor coolant and branch lines out to the block valves are expected to be accessible for inspection with the exception of the welds located within the primary shield. All of these welds will have received a radiographic inspection as part of the fabrication process, and these radiographs will be used for part of the preoperational inspection records. Where possible, an ultrasonic inspection will be performed on all welds in applicable systems for preoperational records and to determine which volumetric inspection techniques will be used in service.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(4.3 & 4.4) Category G - Pressure-Retaining Bolting

All bolting in the piping system is below 2 inches in diameter and is available for visual inspection.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are shown in Table 4.5-7.

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(4.5 & 4.6) Category K - Support Members and Structure for Piping Systems

The piping systems contain supports and restraints whose structural integrity is relied upon to withstand the design loads and seismic induced displacements.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are shown in Table 4.5-7.

(5.0) Pump Pressure Boundary

(5.1 & 5.2) Category L - Pump Casings and Pressure-Containing Welds in Pump Casing

These welds were volumetrically inspected during the pump manufacture, using radiographic techniques. A similar radiographic technique is theoretically possible when the pump is disassembled and the internals have been removed. However, it is not known if such radiograph can be performed in practice due to the radiation level that will probably exist in this component and the geometry of the pump components. A study will be made of the experience with other pressurized-water plant in-service inspections to determine the feasibility of this inspection. Thus, this inspection is not scheduled, as is shown in Table 4.5-7. A visual inspection of the available internal surfaces of the pump will be made at such as the pump is disassembled for maintenance.

The inspections scheduled for the first 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(5.3) Category F - Pressure-Containing Dissimilar Metal Welds

There are no dissimilar metal welds between the reactor coolant piping and the pump.

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(5.4 & 5.5) Category G - Pressure-Retaining Bolting

There are bolts both larger and smaller than 2 inches in diameter on the pumps. If the connection is not broken during the inspection interval, a visual inspection will be made of all bolts and an ultrasonic inspection will be made of bolting 2 inches in diameter and larger with the bolting under tension. If the bolting connection is broken for any reason, the visual and volumetric inspections stated above will be made and, in addition, a volumetric inspection of the ligaments of the base material will be made insofar as is possible due to the cast structure of the pump.

The inspections scheduled for the 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(5.6 & 5.7) Category K - Support Members and Structures for Pumps

The pump contains supports integrally welded to the pressure-containing boundary. All of these supports and restraints will be visually inspected.

The inspections scheduled for the 5-year period and tentatively scheduled for the 10-year period are given in Table 4.5-7.

(6.0) Valve Pressure Boundaries

(6.1) Category M-1 - Pressure-Containing Welds in Valve Bodies

There are no valves in this facility three inches or larger with pressure-containing welds.

(6.3) Category F - Pressure-Containing Dissimilar Metal Welds

There are no dissimilar metal welds between the valves and the piping system in this facility.

(6.4 & 6.5) Category G - Pressure-Retaining Bolting

All valve bolting is below 2 inches in diameter and can be visually inspected in place whether the bolting connection is or is not broken.

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The inspections scheduled for the 5-year period and tentatively scheduled for the 10-year period and are given in Table 4.5-7.

(6.6) Category K-1 - Support Members and Structures for Valves

This facility contains no valves with integrally welded supports.

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Table 4.5-7 - "Components, Parts and Methods of Examination"

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
SECTION A. REACTOR VESSEL AND CLOSURE HEAD						
1.1	A	Longitudinal and circumferential shell welds in core region	Volumetric	None	5% of the length of the circumferential welds and 10% of the length of the longitudinal welds	The required examinations will be made at or near the end of the 10-year inspection interval or when the internals are removed for other reasons.
1.2	B	Longitudinal and circumfer ential welds in shell (other than those of of Category A and C) and meridional and circumferials seam welds in bottom head and closure head (other than those of Category C)	Volumetric	None	5% of the length of the circumferential welds and 10% of the length of the long itudinal welds	The required amount of weld lengths to be examined at or near the end of the 10-year inspection interval or when the internals are removed for other reasons. Excluded are the welds that lie within control element drive mechanism shrouds or the closure head
1.3	C	Vessel-to-flange and head-to-flange circumferential welds	Volumetric	1/3 of the vessel-to-flange and 1/3 of the head-to-flange circumferential weld	Cumulative, 100% coverage of the vessel-to-flange weld	Both of these welds are available for examination during normal refueling operations.

Item numbers refer to Table IS-261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor" Coolant System".

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Table 4.5-7 (Continued)

Item No.	Cate-gory	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
1.4	D	Primary nozzle-to-vessel welds and nozzle-to-vessel inside radiused section	Volumetric	Inspection of two coolant outlet nozzle-to-shell welds and inner nozzle radii	Inspection of all coolant nozzle-to shell welds and inner radius sections	None
1.5	E-1	Vessel penetrations including control element drive penetra tions and control element housing pressure boundary welds	(See Remarks)	(See Remarks)	(See Remarks)	The control element drive mech anism tubes and the in-core penetration nozzles are welded to the upper head with a partial penetration weld. The CEDM assemblies contain an integrally welded thermal sleeve. Volumetric inspection is not possible by currently available techniques, therefore, these penetrations are included in Category E-2.
1.6	E-2	Vessel penetrations including control element drive mechanism penetrations and control element housing boundary welds	Visual	10% of the control element drive mechanism and in-core instru-mentation penetrations will be visually inspected for leakage	Cumulative 25% of the control element drive mechanism and of the in-core instrumentation penetrations will be visually inspected for leakage	None
1.7	F	Primary nozzles-to-safe end welds	Visual and surface and volumetric	The dissimilar metal weld on two coolant out let nozzles	All of the dis similar metal welds on the vessel nozzles will be inspected	The dissimilar metal welds of each nozzle will be inspected at the same time as the nozzle-to-shell weld.

Item numbers refer to Table IS-261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor" Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
1.8	G-1	Closure Studs and nuts	Volumetric and visual or surface	Cumulative 50%	Cumulative 100%	100% visual each year for thread damage.
1.9	G-1	Ligaments between threaded stud holes	Volumetric	1/3 of the vessel-to-flange bolt ligaments will be inspected	Cumulative 100% of the vessel flange bolt ligaments will be inspected	The ligaments will be inspected at the same time as the flange weld of Item No. 1.3.
1.10	G-1	Closure washers, busings	Visual	Cumulative 50%	Cumulative 100%	None
1.11	G-2	Pressure retaining bolting	Visual	10% of the control element drive mechanism and in-core instrumentation penetrations will be visually inspected for leakage	Cumulative 25% of the control element drive mechanism and of the in-core instrumentation penetrations will be visually inspected for leakage	These inspections will be concurrent with those in Item No. 1.6.
1.12	H	Integrally welded vessel supports	Volumetric	(See remarks)	(See remarks)	These welds will be inspected with those of Item 1.4, as allowed by the Code.
1.13	I-1	Closure Head cladding	Visual and surface or volumetric	None	100% of selected areas at or near end of interval	During the 10-year period, at least 6 patches (each 36 square inches) in the vessel head would be inspected.

Item numbers refer to Table IS-261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor" Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
1.14	I-1	Vessel Cladding	Visual	None	100% of selected areas during interval	During the 10-year period, at least 6 patches (each 36 square inches) in the vessel would be inspected.
1.15	N	Interior surfaces and internals and integrally welded internal supports	Visual	A critical examination will be made of the interior surfaces made available by normal refueling operations at the first refueling cycle. This will be repeated at the fourth refueling cycle with the amount of the inspection being dependent upon the results of the first inspection and that made on other pressurized water systems	The inspections made at the fourth refueling cycle will be repeated approximately at the seventh and tenth refueling cycle	The examination will include internal support attachments welded to the vessel whose failure may adversely affect core integrity provided these are available for visual examination by components removed during normal refueling operations.

Item numbers refer to Table IS-261 of the ASME Section XI - "Rules for In-service Inspection for Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
SECTION B. PRESSURIZER						
2.1	B	Longitudinal and circumferential welds	Visual and volumetric	5% of the length of the circumferential welds joining the lower head and of the circumferential weld joining the upper head to the barrel section would be inspected	By the end of the 10-year interval, 10% of the length of the longitudinal and 5% of the length of each circumfer ential weld would be inspected	None
2.2	D	Nozzle-to-vessel welds	Visual and volumetric	The nozzle-to-head welds and inner radii of the surge line would be inspected	The nozzle-to-head welds and inner radii of all nozzles would be inspected	The surge line welds would only be inspected if radiation levels permit.
2.3	E-1	Heater Connections	(See Remarks)	(See remarks)	(See remarks)	All of the penetrations in the pressurizer meet the exclusion criteria of Paragraph IS-120(d).
2.4	E-2	Heater Connections	Visual	15% of the heater connections and two of the instrument and sample penetrations would be visually inspected for leakage	A cumulative total of 25% of the heater connections and the instrument and sample penetrations would be visually inspected for leakage by the end of the interval	None

Item numbers refer to Table IS-261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
---	F	Pressure - containing dissimilar metal welds	Volumetric	100% of the dis similar metal weld joining the surge line connection to the outlet nozzle would be volumetrically inspected by the end of the 5-year interval	100% of the dissimilar metal welds would be inspected	Although this item is not required by Code, it is felt that dissimilar metal welds and stainless steel safe-ends be inspected provided radiation levels permit
2.5	G-1	Pressure-retaining bolting	(See Remarks)	(See Remarks)	(See Remarks)	There is no bolting 2 inches and larger in diameter.
2.6	G-2	Pressure-retaining bolting	Visual	Cumulative 50%	Cumulative 100%	The bolting below 2 inches in diameter would be visually inspected, either in place if the bolted connection is not dis assembled during the inspection interval or whenever the bolted connections are disassembled. The bolting to be inspected would include studs and nuts. Excluded from inspection are bolting of a single connection whose failure results in conditions that satisfy the exclusion criteria of IS-120(d).

Item numbers refer to Table IS-261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
2.7	H	Integrally welded vessel supports	Visual and volumetric	None	The required amount of the support weld would be inspected	None
2.8	I-2	Vessel Cladding	Visual	None	A selected patch of the pressurizer cladding would be inspected by the end of the interval	This will be accomplished only if the pressurizer is opened for another reason.

Item numbers refer to Table IS-261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
SECTION C. STEAM GENERATOR						
3.1	B	Longitudinal and circumferential welds including tube sheet-to-head or shell welds on the primary side	Visual and volumetric	5% of the length of the circumferential weld joining the primary head to the tube sheet of one steam generator would be inspected	By the end of the interval, 5% of the length of all of the welds joining the primary heads to the tube sheets and 10% of the meridional welds would be inspected	None
3.2	D	Primary nozzle to-vessel head welds and nozzle-to-head inside radiused section	Visual and volumetric	The nozzle-to-head welds and inner radii of one generator would be inspected	The nozzle-to-head welds and inner radii of both generators would be inspected	None
3.3	F	Primary nozzle-to-safe end welds	Visual and surface and volumetric	The dissimilar metal welds on one steam generator would be inspected	All of the dissimilar metal welds on the steam generators would be inspected	None
3.4	G-1	Pressure-retaining bolting	(See remarks)	(See remarks)	(See remarks)	There is no bolting 2 inches and larger in diameter.

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
3.5	G-2	Pressure-retaining bolting	Visual	Cumulative 50%	Cumulative 100%	The bolting below 2 inches in diameter would be visually inspected, either in place if the bolted connection is not disassembled during the inspection interval or whenever the bolting connection is disassembled. The bolting to be inspected would include studs and nuts. Excluded from inspection are bolting of a single connection whose failure results in conditions that satisfy the exclusion criteria of Paragraph IS-120(d).
3.6	H	Integrally welded vessel supports	Visual and volumetric	The support welds one steam generator would be inspected	The support welds on both steam generators would be inspected	None
3.7	I-2	Vessel cladding	Visual	A 36-inch-square patch in the heads of one generator would be inspected if a generator is opened and decontaminated for other purposes	A 36-inch-square patch in the heads of all generators would have been inspected by the end of the interval if the generators are opened and decontaminated for other purposes	This will be accomplished only if the steam generator is opened for another reason.

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
SECTION D. PIPING PRESSURE BOUNDARY						
4.1	F	Vessel, pump and valve safe ends-to-primary pipe welds and safe ends in branch piping welds	Visual and surface and volumetric	50% of dissimilar metal welds would be inspected	By the end of the interval, a cumulative 100% of the welds would have been inspected	None
4.2	J	Circumferential and longitudinal pipe welds	Visual and volumetric	15% of the butt welds, including one foot of any longitudinal weld on either side of the butt weld, would be inspected	By the end of the interval, a cumulative 25% of the butt welds in the piping system would have been inspected including one foot of any longitudinal weld on either side of the butt welds	An exception is taken to the welds located within the primary shield.
4.3	G-1	Pressure-retaining bolting	(See remarks)	(See remarks)	(See remarks)	There is no bolting 2 inches and larger diameter in the piping system.

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
4.4	G-2	Pressure-retaining bolting	Visual	50% of the bolting would be inspected	By the end of the interval, 100% of the bolting would be inspected	All bolting is below 2 inches in diameter and would be visually inspected, either in place if the bolted connection is not disassembled during the inspection interval or whenever the bolted connection is disassembled. The bolting to be inspected would include studs and nuts. Excluded from inspection are bolting of a single connection whose failure results in conditions that satisfy the exclusion criteria of Paragraph IS-120(d).
4.5	K-1	Integrally welded supports	Visual and volumetric	15% of the supports (See remarks)	25% of the supports (See remarks)	The integrally welded supports would be inspected only if found to be technically feasible during the preservice examination.
4.6	K-2	Piping support and hanger	Visual	50% of the supports would be inspected	By the end of the interval, a cumulative 100% of the supports would be inspected	The support members and structure subject to inspection would include those supports within the system whose stud integrity is relied upon to withstand the design loads and seismic-induced displacements. The support settings of constant and variable spring-type hangers, snubbers, and absorbers would be inspected to verify proper distribution of design loads among the associated support components.

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
SECTION E. PUMP PRESSURE BOUNDARY						
5.1	L-1	Pump casing welds	Visual and volumetric	(See remarks)	(See remarks)	The only feasible method known to date to volumetrically inspect these pump casing welds is radiography. It is not known if such radiography can be performed in service due to the design and the radiation level in the component. If experience or a study indicates such radiography is possible, the inspection would be performed and a visual inspection would be made, only if the pump is opened for maintenance and the rotating elements are removed.
5.2	L-2	Pump casings	Visual	None	By the end of the interval, a cumulative 100% of the available, inner surfaces of the required pumps would be inspected if the pumps are disassembled for maintenance and the rotating elements are removed	None

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
5.3	F	Nozzle-to-safe end welds	(See remarks)	(See remarks)	(See remarks)	There are no dissimilar metal welds on the pumps.
5.4	G-1	Pressure-retaining bolting	Visual and volumetric	50% of the bolting would be inspected	By the end of the interval, a cumulative 100% of the bolting would be inspected	Bolting 2 inches and larger in diameter would be inspected either in place under tension, or when the bolting is removed, or when the bolting connection is disassembled. The bolting and areas to be inspected would include the studs, nuts, bushings, the holes in the base material, and the flange like segments between threaded stud holes.
5.5	G-2	Pressure-retaining bolting	Visual	50% of the bolting would be inspected	By the end of the interval, a cumulative 100% of the bolting would be inspected	Bolting below 2 inches in diameter would be visually inspected either in place if the bolting connection is not disassembled during the inspection interval, or whenever the connection is disassembled. The bolting to be inspected would include studs and nuts. Excluded from inspection are bolting of a single connection whose failure results in conditions that satisfy the exclusion criteria of Paragraph IS-120(d).

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
5.6	K-1	Integrally welded supports	Visual and volumetric	The support for one pump would be inspected inspected	By the end of the interval, all pump supports would be inspection.	The feasibility of performing the volumetric inspection will be determined during the preoperational inspection. The subsequent in-service inspection will be based on this study.
5.7	K-2	Supports and hangers	Visual	50% of the supports would be inspected	Cumulative 100% of the supports would be inspected	None

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
SECTION F. VALVE PRESSURE BOUNDARY						
6.1	M-1	Valve body welds	(See remarks)	(See remarks)	(See remarks)	Valves with pressure containing welds in valve bodies will be inspected in accordance with the Code.
6.2	M-2	Valve bodies	Visual	50% of the valves required by the Code would be inspected by the Code would be inspected	By the end of the interval, a cumulative 100% of the valves required	None
6.3	F	Valve-to-safe end welds	(See remarks)	(See remarks)	(See remarks)	There are no valves in this system with dissimilar metal welds.
6.4	G-1	Pressure retaining bolting	(See remarks)	(See remarks)	(See remarks)	There are no valves with bolting 2 inches and larger in diameter.
6.5	G-2	Pressure retaining bolting	Visual and volumetric	50% of the bolting would be inspected	By the end of the interval, a cumulative 100% of the bolting would be inspected	All bolting is below 2 inches in diameter and would be visually inspected, either in place if the bolting connection is not disassembled at the inspection interval, or whenever the bolting connection is disassembled. The bolting to be inspected would include studs and nuts.

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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Table 4.5-7 (Continued)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
						Excluded from inspection are bolting of a single connection whose failure results in conditions that satisfy the exclusion criteria of Paragraph IS-120(d).
6.6	K-1	Integrally support	(See remarks)	(See remarks)	(See remarks)	There are no valves with integrally welded supports.
6.7	K-2	Supports and hangers	Visual	50% of the supports and hangers would be inspected	By the end of the interval, a cumulative 100% of the supports and hangers would be inspected	The support members and support subject to inspection would include the supports for piping, valves, and pumps within the system boundary, whose structural integrity is relied upon to with stand the design loads and seismic displacements. The support settings of constant and spring-type hangers, snubbers, and absorbers would be inspected to verify proper distribution of design loads in the associated support components.

Item numbers refer to Table IS261 of the ASME Section XI - "Rules for In-service Inspection of Nuclear Reactor Coolant System".

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4.5.6.3 In-service Inspection for Vital Systems Other Than Nuclear Reactor Coolant Systems

The integrity of vital systems beyond the reactor coolant pressure boundary will be verified by monitoring the performance of the systems during operation and by performing periodic visual inspections for leakage and other signs of abnormal behavior. Any performance anomalies, abnormal leakage and other abnormal operating conditions will be promptly investigated by the operating staff and reported to the Shift Supervisor who is responsible to see that the condition is properly evaluated and that appropriate action is taken. The performance of systems that operated continuously will be routinely observed by the Reactor Operators in the control room or by the Equipment or Auxiliary Operators assigned to the shift. In addition the Equipment and Auxiliary Operators will observe vital equipment in their normal tours through the plant. Vital systems that are not normally operating will be tested periodically as defined by the Technical Specifications. The Technical Specifications define minimum acceptable levels of performance and test frequency. During these periodic tests, the systems will be observed visually for any abnormal behavior.

Should any major repairs or system alterations be required, the system will be tested in accordance with applicable codes and will normally be subjected to hydrostatic and performance tests prior to restoration to service.

4.5.6.4 Precritical Vibration Monitoring Program

The Fort Calhoun precritical vibration monitoring program conforms to the requirements of Safety Guide 20 (Vibration Measurements On Reactor Internals).

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The Fort Calhoun Precritical Vibration Monitoring Program utilizes an instrumentation system designed to monitor the time history of the dynamic response at specified sensor locations (see Figures 4.5-6 and 4.5-7). This instrumentation system consists of twenty-one piezoelectric accelerometers. Eight of the accelerometers are mounted around the circumference of the core support barrel -- five at an axial position corresponding to the thermal shield upper support level, and three at an axial position corresponding to the snubber level. An additional eight accelerometers are mounted around the circumference of the thermal shield -- five at the thermal shield upper support level, and three at the lower jackscrew level. Five external accelerometers are also utilized. Three of these will be attached to the outside of the reactor vessel directly opposite snubber locations.

These accelerometers are magnetically attached to the vessel and can be relocated as required. The two remaining accelerometers are considered roving units and are used to monitor components such as coolant pumps, steam generators, etc. The axis of sensitivity of each sensor is radial. The internal and external accelerometers are monitored simultaneously and, during test conditions, provide the aforementioned time history of the dynamic response.

The internal sensors system is designed to provide sufficient information to define and evaluate, with analytical back-up the "beam" modes of vibration of the core support barrel/thermal shield structural system and the "shell" modes response of this structural system at the thermal shield support level. It is intended that the external sensors system be utilized, on a developmental basis, for in-service vibration monitoring to the extent that it is feasible.

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In addition to an analytical study of the vibrational response of the reactor internals, the Fort Calhoun program includes field measurements of the hydraulic forces exerted on the reactor internals using pressure transducers mounted at five or more selected locations. Strain gages can be added if considered to be necessary. Studies of hydraulic effects included a comparison of measured structural responses with the structural responses predicted analytically using spatially and time dependent hydraulic forcing functions developed by Combustion Engineering. The adequacy of the forcing function predictions have been determined by a comparison with the pressure transducer data.

Data was obtained for the cold and hot pre-core and the cold and hot post-core conditions. The test conditions included one-, two-, three-, and four-pump combinations. Both steady-state and transient flow modes of operation were included.

After the precore testing, the reactor internals were removed and visual or surface examinations conducted to detect any evidence of excessive vibrations, and the presence of wear or flaws induced by unanticipated vibrations.

All time-dependent sensor signals were recorded and the data from the cold pre-core and the cold and hot post-core conditions evaluated. Significant data was further reduced and a spectrum analysis performed. With this reduced data and the theoretical fatigue analysis of the core support barrel-thermal shield system, it was possible to demonstrate the acceptability of the reactor internals design for vibratory loads under normal operating conditions.

A summary of the results obtained from the vibration tests were submitted.

4.5.7 In-service Inspection of Steam Generator Tubes

The surveillance requirements for inspection of the steam generator tubes and tube sleeves ensure the structural integrity of this portion of the RCS will be maintained. The program for in-service inspection of the steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1, dated July 1975. The program for in-serice inspection of steam generator tube sleeves is based on a modification of EPRI PWR Steam Generator Examination Guidelines, Revision 5, dated September 1997. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion. (References 4-15 and 4-22)

4.5.8 NDTT of Other Reactor Coolant System Components

The impact properties of all steel materials which form a part of the pressure boundary of the reactor coolant system were determined in accordance with the requirements of the ASME Code Section III, Paragraph N-330. The materials were required to pass the acceptance test noted in Paragraph N-330 at 40 °F, although it was an objective that the materials meet this requirement at 10 °F. The NDT temperatures of components in the reactor coolant system other than the reactor vessel were determined by Charpy impact tests performed on the carbon steel and alloy steel materials which are part of the pressure boundary. The initial highest NDT temperature for any component in the reactor coolant system was +40 °F for both the steam generators and the pressurizer; this limiting value was determined for the manway cover plates. The operating stress limits for those materials in the reactor coolant system other than the reactor vessel are the same as those for the reactor vessel. Shortly after plant startup, the integrated neutron flux resulted in the reactor vessel being the controlling component.

4.5.9 Nondestructive Tests of Other Reactor Coolant System Components

Prior to and during fabrication of the components of reactor coolant system, nondestructive testing based upon the requirements of Section III of the ASME Boiler and Pressure Vessel Code were used to determine the acceptance criteria for various size flaws. The requirements for the Class A vessels are the same as the reactor vessel. Vessels designated as Class C are fabricated to the standards of Article 21 of Section III of the ASME Boiler and Pressure Vessel Code.

4.5.10 Loose Parts Detection

OPPD has installed a digital loose parts monitoring and analysis system manufactured by ABB/Combustion Engineering. The Vibration and Loose Parts Monitor was custom designed for the Fort Calhoun Station and provides monitoring of the major reactor primary system components in which a loose part could become entrapped. The Loose Parts Alarm Signal activates an annunciator. Operator action is required to reset the alarm.

4.5.11 Shock Suppressors (Snubbers) Operability Requirements

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup or shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of seismic, or other event, initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

4.5.11.1 Applicability

All snubbers required to protect the reactor coolant and other safety related systems shall be operable during Modes 1, 2 and 3 (Operating Modes 4 and 5 for snubbers located on systems required operable in those Operating Modes) except as noted in Steps 4.5.11.2 through 4.5.11.4 below. Applicable snubbers are designated CQE or Limited CQE.

4.5.11.2 Allowed Outage Time for Inoperable Snubbers

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to operable status and perform an engineering evaluation on the supported component or declare the supported system inoperable and follow the appropriate actions specified in the Technical Specifications for that system. When a snubber is found locked up or frozen in place or when a snubber has been inoperable during a seismic event, an engineering evaluation shall be performed, in addition to the determination of the snubber mode of failure. The purpose of the engineering evaluation is to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

Because the snubber protection is required only during low probability events, an inoperable period of 72 hours is allowed for repairs or replacements and an inoperable period of two hours is allowed for surveillance.

4.5.11.3 Allowed Outage Time for Surveillances

A snubber may be removed for surveillance, provided the following conditions are met:

(a) A given snubber station shall not be without an operable snubber for more than two hours during surveillance of attendant snubber. A snubber may be replaced by an operable snubber during surveillance and repair.

(b) No other snubber station is known to be inoperable.

(c) Only one snubber station shall be removed for testing at a time to ensure that no two snubber stations are without an operable snubber during the same time interval.

4.5.11.4 Additions, Changes, and Deletions

Snubbers may be added, changed, or deleted provided an engineering analysis justifies each change.

4.6 Specific References

4-1 Fracture Analysis Diagram Procedure for the Fracture-Safe Engineering Design of Steel Structures, Pellini, W. S. and Puzak, P. P., NRL Report 5920, March 15, 1969.

4-2 "Palisades Primary Coolant Pump Flywheel Fracture Analysis", D. J. Ayres, C-E Report A-69-17-7, July 15, 1969.

4-3 "Finite Element Analysis of Structural Integrity of a Reactor Vessel During Emergency Core Cooling", D. J. Ayres and W. F. Siddall, Jr., C-E Report A-70-19-2, January, 1970.

4-4 "Fracture Mechanics Technology Applied to Heavy Section Steel Structures", E. T. Wessel, W. G. Clark, Jr., and W. H. Pryle, Westinghouse Research Laboratories, November 7, 1968.

4-5 "Interpretive Report of Pressure Vessel Research," T. H. Gross, Weld Research Council Bulletin 101, November, 1964.

4-6 "Effect of Heat Treatment in Microstructure and Low Temperature Property of Pressure Steels", T. H. Gross, E. H. Kottkomp, and R. D. Straut, Welding Journal, Volume 37, 1958.

4-7 Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing". U. S. Nuclear Regulatory Commission, Rev. 2, May, 1976.

4-8 "Allowable Tube Wall Degradation with Modified Support Plate for Omaha Steam Generator", B.H. Boyd, and J. G. Thakkar, Combustion Engineering, Inc., Report Number CENC-1635, April 1984.

4-9 Letter from NRC (A. C. Thadani) to OPPD (R. L. Andrews) dated June 24, 1986.

4-10 Letter from NRC (A. C. Thadani) to OPPD (R. L. Andrews) dated March 25, 1987.

4-11 Letter from NRC (A. C. Thadani) to OPPD (R. L. Andrews) dated April 1, 1987.

4-12 Letter from NRC (S. Bloom) to OPPD (T. L. Patterson) dated March 23, 1994, (TAC M77351 and TAC M77421).

4-13 ABB/CE Calculation A-FC-FE-0002 Rev. 0.

4-14 Letter from OPPD (W. G. Gates) to NRC (Document Control Desk), Dated October 15, 1993 (LIC-93-0258).

4-15 NRC Safety Evaluation Supporting Amendment No. 46, Steam Generator Tube Inspection Program, July 2, 1979.

4-16 PWR Primary Water Chemistry Guidelines, Electric Power Research Institute, R. A. Shaw, Report Number NP-4762-SR

4-17 NRC Safety Evaluation Supporting Amendment No. 142, March 19, 1992.

4-18 NRC Safety Evaluation Supporting Amendment No. 176, September 27, 1996.

4-19 EA-FC-96-002, Rev. 1, Revision of TSP (Tri-Sodium Phosphate Dodecahydrate) Quantity in the Containment Sump.

4-20 Oak Ridge National Laboratory Contract No. W-7405-ENG-26, "Design Considerations of Reactor Containment Spray Systems - Part X The Stress Corrosion Cracking of Types 304 and 316 Stainless Steel in Boric Acid Solutions", May 1971.

4-21 NRC Safety Evaluation Report Supporting Amendment No. 179, December 30, 1996.

4-22 NRC Safety Evaluation Report Supporting Amendment No. 195, March 1, 2001

4-23 EA-FC-01-011, Rev. 0 "Low Temperature Overpressure Protection for 20 EFPY."

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SECTION 5

STRUCTURES

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5.4 CONTAINMENT LOADINGS

The loadings discussed in the following subsections were considered in the structural analysis of the containment structure.

5.4.1 Dead Load

Dead load consists of the weights of foundation mat, cylindrical wall, domed roof, internal concrete and equipment.

Equipment was included in the dead load since all major equipment is fixed at given locations and is subject to negligible variations in weight from the initial values. Appropriate impact and dynamic loadings were assigned to equipment for the structural design of their supports. However, since all equipment is supported by the internal concrete these loadings were not included in the containment structure analysis coincident with accident loadings. Total equipment loads form a small part of the total dead load of the containment structure.

The polar crane runway is supported from the cylindrical wall. The appropriate static and dynamic loadings from the crane were considered in the containment analysis (see also Section F.2.2.5 of Appendix F).

5.4.2 Live Load

Live load consists of the snow load on the domed roof of the structure and the assumed floor loadings on the internal concrete located within the containment.

The design snow load was 30 psf of horizontal projection of the roof. Floor loadings were established in accordance with their intended use.

5.4.3 Design Pressure

The design pressure was 60 psig based on the design basis accident (DBA) pressure as discussed in Section 14.16.

5.4.4 Thermal Loads

Thermal loads, based on a maximum normal operating temperature of 120°F and on a design accident temperature of 305°F (which is in excess of the DBA temperature), were evaluated from the temperature gradients developed through the containment shell.

The maximum temperature differential across the containment walls and dome is 110.5°F for the non-accident condition. This gradient was computed using the maximum design containment ambient temperature of 120°F and the lowest recorded temperature at Eppley Airfield of -22°F (see Notes on Table 2.5-5).

The transient thermal gradients through the containment envelope during the DBA are shown in Figure 5.4-1. The DBA temperature gradients were developed using the CONTRANS Code as described in Section 14.16. Initial temperatures in the containment wall were those which occur in the winter. This gives the initial thermal gradient.

It can be seen from Figure 5.4-1 that the post-accident thermal gradients in the concrete are not significantly more severe than the operating temperature gradient for the winter extreme. The zones of increased temperature within the concrete wall were extremely localized initially, and as these zones increase with time the internal pressure within the containment reduces drastically. The temperature of the liner is strongly affected by the rise in temperature of the containment atmosphere. However, under the combined loading of prestress, dead load, normal operating temperature (winter extreme), and concrete shrinkage and creep, the liner is already stressed to a value approaching the yield strength of the steel. Thus accident temperatures would increase the liner stress by only a negligible amount.

In comparison, USAR Section 14.16 presents the containment pressure analysis and includes a short duration temperature excursion (100 seconds) above the nominal 305°F post accident transient temperature presented in Section 5.4. The statements made in these respective USAR Sections need to be understood within the context of each section. The objective of USAR 14.16 is to demonstrate that containment pressure will not exceed 60 psig. USAR Section 5.4 discusses containment loadings during an accident transient and the objective with respect to the design containment accident temperature (305°F) is to show that the containment structure is not significantly affected, temperature wise, during the DBA. Therefore, a very short duration temperature excursion above the 305°F containment structure design criterion temperature would have a negligible affect on both the containment liner and the massive containment structure itself; and does not affect the thermal loading analysis of the containment structure.

5.4.5 Design Exterior Pressure

The design exterior pressure was 2.5 psi. This is the positive differential pressure between the outside and the inside of the containment and would be realized under the following sequence of atmospheric and operational events:

a. The containment structure is sealed while the internal temperature is 120°F and the external barometric pressure is 29.0 inches of mercury;

b. The containment is then cooled so that the internal temperature becomes 80°F with a simultaneous increase in external barometric pressure to 31.0 inches of mercury.

5.4.6 Wind Load

The wind load was based on the recommendations of ASCE Paper 3269, "Wind Forces on Structures." The fastest mile of wind at the site location for a 100 year period of recurrence is a 90 mph basic wind at 30 feet above ground level. Shape and gust factors and wind velocity variations with heights were employed from the same reference. Containment structure wind loading diagrams are shown in Figure 5.4-2.

5.4.7 Tornado Load

Definitive data regarding loadings actually experienced during tornadoes was not available; this lack of information was primarily due to the destruction of recording instruments at the time of maximum wind velocities.

It is generally recognized that well designed and constructed conventional structures withstand tornadoes with relatively minor damage. Reinforced concrete structures seem to suffer the least damage when compared with other types of construction. Where damage has occurred, the primary factor responsible appears to have been the explosive release of air pressure within the building when the low atmospheric pressure within the tornado vortex suddenly enveloped the structure. The containment structure, designed to withstand a 60 psig internal pressure resulting from an internal accident, is inherently safe against this type of loading.

A possible associated effect of tornadoes upon structures is that due to impact of tornado-borne material. Various items such as debris or portions of demolished structures may be picked up by the tornado and propelled at considerable velocity against any structure in their path. Therefore several such tornado-borne missiles were postulated and their effect on the containment structure evaluated.

Based on studies of tornado damage, the peripheral wind velocities are frequently estimated in excess of 300 mph. An "average" tornado is thought to have peripheral wind velocities in the range of 200 mph. The maximum value of pressure below atmospheric at the center of the vortex is estimated as 3 psi.

The containment structure was designed to maintain its structural integrity and thus permit a safe shutdown in a tornado with a maximum wind velocity of 500 miles per hour. A concurrent pressure drop of 3 psi applied in a period of 3 seconds was assumed as the tornado passes across the structure.

In addition, the containment structure can withstand the torsional moment resulting from the drag of peripheral winds of 500 mph at the entire surface of the cylindrical wall exterior.

The containment shell is also resistant against the impact effect of hypothetical tornado-borne missiles as discussed in Section 5.8.2.2.

5.4.8 Seismic Loads

Seismic loads for the containment were based on a design earthquake and a larger maximum hypothetical earthquake as discussed in Appendix F. The simultaneous ground accelerations were:

a. Design earthquake: 0.08g horizontal and 0.053g vertical;

b. Maximum hypothetical earthquake: 0.17g horizontal and 0.113g vertical.

The corresponding loadings used for containment design were determined by a dynamic analysis.

Seismic instrumentation is provided as discussed in Appendix F.

5.4.9 Hydrostatic Load

The containment design includes the effect of external hydrostatic loads resulting from variations of ground water level from a low of elevation 980 feet to a maximum flood level of elevation 1014 feet (see Section 2.7.1.2).

NT SECTION 6

ENGINEERED SAFEGUARD SYSTEMS

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The following figures are controlled drawings and can be viewed and printed from the applicable listed aperture card.

6. ENGINEERED SAFEGUARDS

6.1 GENERAL

6.1.1 Definition and Function

Engineered safeguards is the designation given to systems and components provided to protect the public and plant personnel by minimizing both the extent and the effects of an accidental release of radioactive fission products from the reactor coolant system, particularly those following a loss-of-coolant accident up to and including a double ended rupture of the largest reactor coolant pipe. These safeguards function to localize, control, mitigate, and terminate such accidents and to hold off-site environmental exposure levels within the guidelines of 10 CFR Part 100.

The systems function to cool the core, limit the magnitude and duration of the pressure transient within the containment vessel following a loss-of-coolant accident, and provide long term post-accident cooling. Such an accident and a gross release of fission products could occur only as the result of an incredible series of failures and malfunctions. The spectrum of loss-of-coolant accidents which could result from piping failure and the offsite consequences thereof are defined and analyzed in Section 14.15. In addition, the performance of the engineered safeguards is relevant to the analysis required by 10 CFR Part 100.11.

6.1.2 System Descriptions

Engineered Safeguards Equipment include Engineered Safety Features Systems, Essential Auxiliary Support Systems and Engineered Safeguards Controls and Instrumentation.

6.1.2.1 Engineered Safety Features Systems

a. Safety Injection System

The safety injection system injects borated water into the reactor coolant system. This provides core cooling to limit damage and fission product release, and ensures adequate shutdown margin regardless of temperature. The injection system also provides continuous long term post-accident cooling of the core by recirculation of borated water from the containment recirculation line inlet at elevation 994'-0". The safety injection system includes the safety injection tanks, high-pressure safety injection pumps and low-pressure safety injection pumps.

b. Containment Spray System.

The containment spray system removes heat by spraying cool borated water through the containment atmosphere. Heat is transferred to the component cooling system through the shutdown heat exchangers.

c. Containment Air Cooling and Filtering System.

This system removes heat by circulating the post-accident containment atmosphere over coils cooled by the component cooling water system and removes particulates by filtration.

d. Containment Hydrogen Purge System

The containment hydrogen purge system is designed to provide a safe, independent, monitored, and controlled means of purging any potential accumulation of hydrogen in the containment. This prevents the hydrogen concentration in the containment from exceeding 3 percent (by volume) following a loss of coolant accident (LOCA). See Section 9.10 for a description of the containment hydrogen purge system.

e. Auxiliary Feedwater System

The auxiliary feedwater system serves to supply feedwater to the steam generators whenever the reactor coolant temperature is above 300°F and the main feedwater system is not in operation, e.g. during startup, cooldown or emergency conditions resulting in a loss of main feedwater. This system is described in Section 9.4.

f. Containment Isolation System

The containment isolation system consists of containment isolation valves and dampers actuated by the containment isolation actuation signal (CIAS) as described in Sections 5.9 and 7.3.

6.1.2.2 Essential Auxiliary Support Systems

a. Normal Station Electrical Power Distribution System

Portions of the normal station distribution system that provide power to engineered safety features systems and essential auxiliary support systems and components is an essential auxiliary support system. These essential components and equipment in the normal station electric distribution system are described in Section 8.3. Reference: USAR FIG 8.1-1, Simplified one-line diagram, Plant Electrical System.

b. Emergency Power Systems

The emergency power systems including station batteries and the emergency diesel generator systems, together with their fuel storage and fuel transfer systems are essential auxiliary support systems. These systems are described in Sections 8.1, 8.2, and 8.4. Reference: USAR FIG 8.1-1 Simplified One-Line Diagram, Plant Electrical System.

c. Instrument Air System

Those portions of the instrument air systems i.e. check valves, accumulators, components and tubing downstream of the check valves, that are required to operate air operated valves in the event of a design bases accident (DBA) are essential auxiliary support system components. These components are described in Section 9.12.

d. Auxiliary Building HVAC System

Those portions of the auxiliary building HVAC system which are necessary for exhaust of the containment hydrogen purge system are essential auxiliary support system components. These components are described in Section 9.10.

e. Control Room HVAC System

HVAC systems which are required for operation of engineered safeguards are considered to be essential auxiliary support systems. The control room HVAC system, which fulfills this function, is described in Section 9.10.

f. Component Cooling Water System

Portions of the component cooling water system provide cooling water necessary for operation of engineered safeguards equipment. The component cooling water system is described in Section 9.7.

g. Raw Water System

The raw water system provides a cooling medium for the component cooling water system. The raw water system is described in Section 9.8.

6.1.2.3 Other Systems Actuated by Engineered Safeguards Signals

The chemical and volume control system (CVCS) operates in conjunction with the safety injection system to inject concentrated boric acid into the reactor coolant system on receipt of a pressurizer pressure low signal (PPLS) and/or a containment pressure high signal (CPHS). Because this system is not necessary to mitigate the consequences of accidents, as documented in USAR Section 14, this system including charging pumps is not classified as Engineered Safeguards equipment. The operation of the CVCS is described in detail in Section 9.2.

6.1.2.4 Engineered Safeguards Controls and Instrumentation

Engineered safeguards controls and instrumentation includes all instruments, instrument loops, and control loops necessary to monitor and control all aspects of the engineered safety features systems and the essential auxiliary support systems. This equipment is described in Sections 7.3 and 7.6.

6.1.3 System Design and Reliability

Environmental qualification of electrical equipment for harsh environment is discussed in Section 1.6. The following discussion was the original design criteria and may differ from the above noted discussion which is the present day design criteria.

The engineered safeguards system components were procured to detailed engineering specifications and tested to applicable codes. In addition, the special testing of certain actual or prototype components under the limiting service conditions was required by the equipment specifications (see Section 1.4). The double-ended rupture of the largest reactor coolant pipe was designated as the design basis accident (DBA), since the forces and thermal phenomena affecting the core were the most severe. A discussion of the analyses performed for a spectrum of break sizes, to determine these limiting service conditions, is presented in Section 14.15.

Components of the engineered safeguards systems and associated critical instrumentation essential for operation following a DBA were designed and tested to operate in the environment to which they would be exposed in this event. These design requirements were of primary significance for those portions of the engineered safeguards system which are located inside the containment structure, including filters, cooling coils, instrumentation, electrical wiring and motors.

The forces generated by the maximum hypothetical earthquake (see Appendix F), combined with the rupture of a reactor coolant pipe, were considered in the design of the engineered safeguards. The design assures that the functional capability of the systems would be retained. Vessels which are connected to the engineered safeguards systems are supported and restrained to allow controlled movement during this load condition, and piping was designed to accept the imposed movements. Engineering calculations of the flexibility of the systems were performed to verify that the piping can accept these additional vessel movements and still remain within code allowable limits of stress. Flexibility calculations were performed according to the Code of Pressure Piping, USAS B31.7.

For the Fort Calhoun Station, piping 2" and smaller was field fabricated, while piping 2 1/2" and larger was shop fabricated. The Engineer provided dimensioned drawings routing shop fabricated piping which the piping fabricator was required to follow. Piping in the 2" and smaller region was routed but not necessarily dimensioned on the engineer's drawings. While the piping erector was required to follow the routing indicated on the drawing, he was free to make local modifications as necessary to avoid interferences. The engineered safeguards systems were identified to the Construction Management group, and they were required to identify any modifications in routing to the Engineer's design group. The design group then verified that the changes in routing did not cause the pressure drop in the system to exceed allowable limits. The field fabricated piping in the emergency core cooling system is that indicated as 2" and under on P&ID E-23866-210-130. Piping in this category which is in the direct path from pumps to the reactor coolant system are some branches of the high pressure safety injection system. During preoperational testing, the flow rate through the high pressure safety injection system was verified using the flow elements provided in the piping.

The specification values for radiation exposure were estimated considering the equipment location within the plant. Where applicable the exposure resulting from the Maximum Hypothetical Accident was used to establish the specified value. In all cases a value considerably in excess of the exposure expected at the parts of the equipment susceptible to radiation damage was used in the equipment specifications.

The metallic materials of construction of the equipment show no measurable changes in properties for radiation exposures many orders of magnitude greater than specified.

Identification and qualification of electrical equipment and instrumentation was done in accordance with the EEQ program described in Section 1.6.16.

6.1.4 Seismic Design Evaluation

The reactor protective system and engineered safeguards components were designed to meet Class I seismic criteria as delineated in Appendix F of the USAR.

The ability of this equipment to perform under Class I seismic criteria has been verified either by shop test, prototype test, field test, or seismic analysis prior to plant operation; an exception to these seismic requirements exists in the case of most pumps and compressors since their normal performance design criteria with respect to vibration and shock loading exceed these seismic requirements. Raw water pumps were analyzed for seismic response due to their importance in an accident situation and their vertical pump shaft configuration.

6.3 CONTAINMENT SPRAY SYSTEM

6.3.1 Design Bases

The function of the containment spray system is to limit the containment pressure rise and reduce the leakage of airborne radioactivity from the containment by providing a means for cooling the containment following a loss-of-coolant accident (LOCA). This system reduces the leakage of airborne radioactivity by effectively removing radioactive particulates from the containment atmosphere.

Pressure reduction is accomplished by spraying cool, borated water into the containment atmosphere which provides a means for cooling the containment atmosphere. Heat removal is accomplished by recirculating and cooling the water through the shutdown heat exchangers. The system is independent of the containment air cooling and filtering system described in Section 6.4 for the containment pressure analysis described in Section 14.16.

Removal of radioactive particulates is accomplished by spraying water into the containment atmosphere. The particulates become attached to the water droplets which fall to the floor and are washed into containment sump.

All system components were designed to withstand Seismic Class 1 loadings (see Appendix F).

6.3.2 System Description

The system consists of the Safety Injection and Refueling Water Tank (SIRWT), three spray pumps, two heat exchangers (shutdown cooling heat exchangers) and all necessary piping, valves, instruments and accessories. The pumps discharge the borated water through the two heat exchangers, during recirculation, to a dual set of spray headers and spray nozzles in the containment. These spray headers are supported from the containment roof and the spray nozzles are arranged in the headers to give essentially complete spray coverage of the containment horizontal cross section area. One pump meets the capacity requirements in the event of a DBA.

Two spray pumps are located in one engineered safeguards room, along with one HP and one LP injection pump. The third spray pump is located in the second engineered safeguards room with one LP and two HP pumps. Both engineered safeguards rooms are located below grade at elevation 971'-0" in the auxiliary building. The shutdown cooling heat exchangers are located in two rooms of the auxiliary building at elevation 989'-0".

Each engineered safeguards room has a separate pump suction from both the SIRWT and the containment recirculation line inlet to ensure that the pumps in one room will have adequate suction if the suction line to the second room fails. The containment spray system is shown in P&ID E-23866-210-130.

6.3.3 System Components

Ratings of the equipment are given in Tables 6.3-1 and 6.3-2. Additional detail will be found in Tables 6.2-1 and 6.2-3. The design of the spray pump casing is identical to the low-pressure safety injection pumps described in Section 6.2.3.2.

Table 6.3-1 - "Containment Spray System Component Performance"

Containment Spray Pumps, Item No's. SI-3A, 3B and 3C

Number of Units	3
Motor Nameplate Voltage	460
Horsepower, hp	300
Pump Design Point Flow, gpm	1700
Total Head at Design Point Flow, ft	450

(See Section 6.2.1 for NPSH discussion).

Shutdown Heat Exchangers, Item No's. AC-4A and 4B

Number of Units	2
Capacity (each) component	58.9×10^6 Btu/hr based on 2,937gpm of cooling water at 95°F inlet temperature and 2,250gpm of spray water at 212°F inlet temperature

Table 6.3-2 - "Summary of Piping, Valve and Spray Nozzle Characteristics"

Code	USAS B31.7 1968, Class II
Material	
Valves & Piping	304 Stainless Steel
Design Temperature, °F	35 0
Design Pressures, psig	
Piping, Suction	66
Piping, Discharge*	500
Piping and Valve Construction	
2-1/2 in. and larger	Butt welded, except at flanged equipment
2 in. and smaller	Socket welded, except at screwed or flanged equipment
Spray nozzles	
Type	Hollow cone, centrifugal, w/vanes
Number, per spray header	274 (264 minimum operable)
Flow characteristic	12.4 gpm @ 42 psid
Spray droplet size, mean, microns	1800

*Includes piping between pump discharge and spray header AOVs HCV-344/345.

6.3.4 System Operation

6.3.4.1 Normal and Shutdown Operation

During periods of normal or shutdown plant operation, the spray system is normally not in service. Under certain limited plant shutdown conditions as described in Section 9.3.6, the containment spray pumps can be considered as available shutdown cooling pumps.

6.3.4.2 Emergency Operation

All three spray pumps are started by the containment spray actuation signal (CSAS) via the sequencers. The containment spray actuation signal (CSAS) brings the system to full operation (see Section 7.3). If all normal power sources are lost and one emergency diesel-generator fails to start, at least one spray pump is started via the sequencers. One spray header valve will open, with the second spray header valve opening only if SI-3B and SI-3C start.

Initially, the pumps take suction from the SIRWT. Upon reaching low tank level the recirculation actuation signal (RAS) is initiated, automatically transferring the pump suction to the containment recirculation line inlet at elevation 994'-0". The recirculated water is cooled by component cooling water in the shutdown heat exchangers prior to discharge into the containment atmosphere. During the recirculation phase a portion of the cooled effluent from the shutdown heat exchangers may be directed to the suction of the high-pressure safety injection pumps. This connection to the high-pressure injection pump suction is provided with a normally closed, fail-closed, remote manually operated open-shut valve.

6.3.5 Design Evaluation

The containment spray system is designed for a heat removal capacity that is sufficient to maintain the peak containment pressure below the design limit as discussed in Section 14.16.

The iodine removal capability during the first 30 days of the DBA (large break LOCA) is discussed in Section 14.15.

The minimum required hydraulic performance for a containment spray pump is calculated based on the credited containment spray flow in the LOCA containment pressure analysis for the one-pump, one-header operating mode (Ref. 14.16-6).

The assumed SIRWT temperature and credited flow of the containment spray system have been established through development of inputs for the containment pressure analysis, documented in Section 14.16. The nozzles are designed to discharge spray droplets with a mean diameter of less than 1800 microns. It has been shown by analysis that all of the spray will be essentially in thermal equilibrium with the containment atmosphere before reaching the collected containment water.

6.3.6 Availability and Reliability

The spray pump pressure-containing parts were hydrotested at 1.5 times the design pressure. At design temperature, the pressure rating of the pump suction and discharge piping is at least 3 times the maximum expected operating pressure.

Upon depletion of the SIRWT, the containment spray pumps take suction from the containment recirculation line inlet and discharge through the two heat exchangers to the containment spray headers. The discharge from the containment spray pumps is piped into the containment building and into each of the duplicate spray headers.

System availability is enhanced by the separate suction headers from the SIRWT and the containment recirculation line inlet, by the provision of two shutdown cooling heat exchangers, and by the fact that the low-pressure safety injection pumps are available for this service in the recirculation mode.

The safety injection and containment spray pumps are considered to be operable in accordance with Technical Specification 2.3 and 2.4 in the event component cooling water is not available to cool the pump seals and bearings. This conclusion is based on Engineering Analysis EA-FC-91-014. This condition does not apply to shutdown cooling operation in accordance with Technical Specification 2.1.1 or 2.8.

6.3.7 Tests and Inspections

The spray pumps and heat exchangers are located outside the containment to permit access for periodic testing and maintenance during normal plant operation.

A recirculation line is provided on the discharge of each spray pump. Periodic testing is performed by recirculating water back to the SIRWT. The recirculation line is sized to pass the minimum allowable pump flow.

The three identical spray pumps were shop tested at sufficient head capacity points to generate complete performance curves. NPSH requirements for the capacity range were verified by a suction pressure suppression test for each pump. A shop thermal transient test from 50°F to 300°F, performed on one of the identical low-pressure injection pumps (see Section 6.2), assured that the design was suitable for the switch over from the injection to the recirculation mode. Further information on pump testing is given in Section 1.6.

Performance data for one spray nozzle were provided which show manufacturing tolerances such that the maximum spray droplet mean diameter will not exceed 1800 microns at design conditions.

6.4 CONTAINMENT AIR COOLING AND FILTERING SYSTEM

6.4.1 Design Bases

6.4.1.1 General

The containment air cooling and filtering system was designed to limit the leakage of airborne activity from the containment in the event of a loss-of-coolant accident. This is accomplished by:

a. The removal of heat released to the containment atmosphere during the Design Basis Accident (DBA) to the extent necessary to initially maintain that structure below the design pressure and then reduce the pressure to near atmospheric. Leakage from the containment is thereby restricted to within design limits.

b. The prevention of the accumulation of hydrogen pockets by maintaining a continuous flow throughout the containment. The minimum number of air changes in restricted areas of the containment is one per hour which provides adequate mixing and sweeping of hydrogen. A lesser number of air changes is permitted inside the fuel transfer canal, steam generator cells, pump cells, and pressurizer cell as these are open top cells in which hydrogen will not tend to accumulate.

The system also functions during normal plant operation, outside the context of engineered safeguards, to cool the containment atmosphere and provide any filtration that may be required prior to personnel access (see Section 9.10). The system is independent of the containment spray system.

6.4.1.2 Design Criteria and Performance Objectives

The heat removal capability of the system is based upon the design basis accident (DBA). The system was designed to remove heat from air saturated with moisture at a design pressure of 60 psig and temperature of 288°F to maintain the containment below its design pressure of 60 psig. A heat removal rate of 280 x 106 Btu/hr maximum capacity with IA available, is necessary to meet this requirement. The containment pressure transient and the capability of the heat removal systems in maintaining the transient below the design limit are discussed in Section 14.16.

The criteria for system and equipment design and performance include the following:

a. Ability to withstand a pressure increase from atmospheric to 60 psig in seven seconds with unimpaired function at the commencement of the DBA;

b. Ability to operate for an extended period in an atmosphere of air saturated with borated water at 60 psig and 288°F;

c. Resistance to seismic effects without impaired function (Class I design, see Appendix F);

d. Protection from missiles;

e. Fail-safe design of active components as far as is practicable;

f. Reliable initiation of the required operational sequences including transfer to alternate power sources (see Section 7.3 and 8.4);

g. Facility for the periodic inspection and testing of the operability and functional performance of components.

6.4.2 System Description

The system consists of four air handling units, each with its own fan, a common plenum discharge system and instrumentation and controls. There are two types of units; two have filtering capacity and the other two have no filtering capacity. The arrangement of the equipment is shown in Figure 6.4-1.

The air cooling and filtering units comprise, in flow sequence, inlet face dampers, baffle type moisture separators, media type mist eliminators, HEPA filters, charcoal filters and cooling coils, all contained in a single housing. Dampers between the charcoal filters and the cooling coils allow the filter banks to be bypassed during normal (i.e., non-accident) operation. The filter banks of each unit are split in two parallel and separate trains. The common exhaust flows from each train are drawn through coil banks by axial, air-over-motor fans and discharged into a plenum. Backdraft dampers are installed in the duct sections downstream of the fans. The arrangement of the units on the platform at elevation 1060'-0" is shown in Figure 6.4-2. Each unit was designed for an inlet air flow of 110,000 CFM when cooling the containment atmosphere at the DBA conditions of 60 psig, 288°F and 100 percent relative humidity to remove 140 x 10^6 Btu/hr.

The air cooling units are similar in design to the cooling and filtering units but do not include mist eliminators, face and bypass dampers, HEPA filters and charcoal filters. The moisture separators and cooling coils of each unit are arranged in a single flow train as shown in Figure 6.4-3. They are located on the operating floor at elevation 1045'-0". Each unit was designed for an inlet air flow of 66,000 CFM to remove 70 x 10^6 Btu/hr at DBA conditions.

During normal operation cooled air is discharged from the plenum through a duct system to those areas where cooling is required (see P&ID 1405-M-1). At the DBA the plenum discharge is made through hatches, at the lower end of the plenum, designed to open on a temperature increase. In addition, a 24" diameter damper also designed to open on a temperature increase, is located at the top of the discharge plenum to distribute air to the containment dome after a DBA.

The design values for the air handling units are shown in Table 6.4-1, and non-accident data are also shown for comparison. The DBA data are design capacity values. The signal initiating emergency safeguards (see Section 7.3.2) brings all four units into operation. The expected heat removal by the containment air coolers will meet or exceed that credited in the containment pressure analysis for the applicable accidents. Currently, a contribution from the containment air coolers is credited in the mitigation of peak containment pressure for a Main Steam Line Break, but not for a LOCA.

The system is normally manually operated from the control room but in the event of a loss-of-coolant accident it is automatically brought to the emergency operating condition. Critical temperatures and differential pressures across filter banks, cooling coils and fans are continuously indicated in the control room with appropriate alarms where necessary.

The containment air cooling and filtering system is dependent upon the component cooling water, raw water and electrical systems which are discussed in Section 9.7, 9.8, and 8 respectively.

Table 6.4-1 "Containment Air Cooling and Filtering System"
Air Handling Units and Cooling Coils Data

NOTE: The values shown are "spec sheet" data relevant to the original sizing of the units. This data does not necessarily reflect actual operating parameters for either post-accident or normal operating conditions.

	Cooling & Filtering Units (2 each)		Cooling Units (2 each)	
	DBA Operation	Normal Operation	DBA Operation	Normal Operation
Heat Removal Capacity per unit	140 x 10^6 Btu/hr	2.46 X 10^6 Btu/hr	70 x 10^6 Btu/hr	1.23 x 10^6 Btu/hr
Medium Handled	Air/Sat. Steam Mix	Air	Air/Sat. Steam Mix	Air
Containment Pressure	60 psig	0 psig	60 psig	0 psig
Air Side Entering Flow per unit	110,000 CFM	94,000 CFM	66,000 CFM	52,000 CFM
Air Side Leaving Flow per unit	86,500 CFM	90,000 CFM	52,000 CFM	50,000 CFM
Pressure Drop	1.0 in H_2O	0.9 in H_2O	1.5 in H_2O	1.3 in H_2O
Face Velocity	582 ft/min	497 ft/min	750 ft/min	590 ft/min
Air Side Entering Density	0.201 lb/ft^3	0.068 lb/ft^3	0.201 lb/ft^3	0.068 lb/ft^3
Air Side Leaving Density	0.217 lb/ft^3	0.071 lb/ft^3	0.213 lb/ft^3	0.071 lb/ft^3
Air Side Inlet Temperature	288°F	120°F	288°F	120°F
Air Side Outlet Temperature	271°F	98°F	274°F	100°F
Water Vapor Condensation Rate per unit	150,000 lb/hr	N/A	75,000 lb/hr	N/A
Containment Volume Changes per hour	6.3 per unit	5.4 per unit	3.7 per unit	3.0 per unit
Cooling Water Flow per unit	2340 gpm	450 gpm	1170 gpm	255 gpm
Pressure Drop	14.4 ft H_2O	0.9 ft H_2O	18.0 ft H_2O	1.0 ft H_2O
Tube Velocity	4.4 ft/sec	0.8 ft/sec	5.6 ft/sec	1.0 ft/sec
Cooling Water Inlet Temperature	120°F	90°F	120°F	90°F
Cooling Water Outlet Temperature	240°F	101°F	240°F	101°F
Fouling Factor (ft^2-hr-°F/Btu)	0.0005	0.0005	0.0005	0.0005

6.4.3 System Components

6.4.3.1 Dampers

The dampers on the cooling and filtering units are of multi-blade construction with galvanized steel blades and neoprene seals. The air piston operators are fail safe; the face dampers open and the bypass dampers close on loss of air pressure or control signal.

6.4.3.2 Moisture Separators and Mist Eliminators

The cooling and filtering units moisture separators and mist eliminators protect the HEPA filters, which are immediately downstream, from water droplet impingement damage and blockage ("blinding"). In the cooling units the separators protect the cooling coils from impingement damage and from water loading which would increase flow resistance and impair heat transfer.

The separators consist of inlet moisture separating baffles, designed to remove large entrained water droplets by impingement and subsequent trapping. The mist eliminator cells each contain three removable, 2 inch thick glass fiber pads held by wire retainer grids. The cells are 24 inches square in face dimension and the joints between cells are sealed to prevent air bypass. Each cooling and filtering unit contains 96 cells. Moisture removed at the baffles and cells is collected in horizontal tiers of drain troughs and is allowed to cascade, out of the air stream, into a drained sump. All materials are corrosion and fire resistant.

The performance data for the combined moisture separator and mist eliminator units are presented in Table 6.4-2.

Table 6.4-2 - "Moisture Separator/Mist Eliminator Performance Data"

Rated Flow Per Cell, CFM	1530
Removal Efficiency on Water Particles >1 Micron Diameter, % by weight	99.0
Design Pressure Drop at Rated Flow, in. H_2O	1.1 to 1.5

6.4.3.3 HEPA Filters

The high efficiency particulate air (HEPA) filters are located upstream of the charcoal filters to prevent the latter from becoming loaded with particulates which would reduce their efficiency.

The HEPA filter banks consist of individual filter cells 24 inches wide by 24 inches high by 12 inches deep supported by a holding frame. The cell casings are of cadmium plated steel construction. The filter medium is pleated fiber glass separated by aluminum spacers and is suitable for the DBA environmental operating conditions. The filters are removable and are retained in the holding frame by latches. Cell-to-frame flanges and gaskets prevent air bypass. All materials are corrosion and fire resistant. The filters meet the requirements of Military Specification MIL-F-51068A, "Filter, Particulate, High Efficiency, Fire Resistant" and USAEC Health and Safety Information Bulletin, Issue No. 212, June 15, 1965.

Each cooling and filtering unit contains 96 HEPA filter cells. Filter performance data are presented in Table 6.4-3.

Table 6.4-3 - "HEPA Filter Performance Data"

Rated Flow per Cell, CFM	1145
Removal Efficiency on Particles >0.3 Micron Diameter, % by count	99.97
Design Pressure Drop at Rated Flow, in. H_2O	1.0 to 2.0

6.4.3.4 Charcoal Filters

The charcoal filter banks consist of individual filter cells approximately 24 inches wide by 6 inches high by 26 inches deep supported by a holding frame. Each filter cell contains two horizontal adsorber beds, each 2 inches thick, of activated, dust-free, charcoal. The air flow path is baffled for horizontal intake and discharge. The cell frames and perforated charcoal bed screens are stainless steel. The cells are arranged in groups of three in removable filter frames. These filter frames are retained in the housing holding frames by screwed clamps; cell-to-frame flanges and gaskets prevent air bypass.

Each cooling and filtering unit contains 288 charcoal filter cells.

Table 6.4-4 - "Charcoal Filter Performance Data"

Rated Flow per Cell, CFM	383
Pressure Drop, in. H_20	1.1

6.4.3.5 Cooling Coils

The cooling coils are of the finned tube, double serpentine type. The tubes are of copper and are 5/8 inch O.D. with 0.022 inch wall thickness. The plate type aluminum fins are mechanically bonded to the tubes. The tubes are oriented for horizontal water flow and each bank is 12 rows deep. A galvanized steel casing supports the tube bank. Removable plugs in the headers are provided to permit tube cleaning. The contract specification for the coils required that they be pressure tested at 250 psig with air under water. This test pressure exceeds the hydrostatic test pressure on the piping system to which the coils are connected, thereby demonstrating that the pressure retention capability of the coils is adequate for the intended service. The coils can withstand an external pressure considerably in excess of 60 psig without collapse. Each cooling and filtering unit incorporates 21 coils, each 54 inches tube length by 24 inches (16 tubes) wide. Each cooling unit incorporates 8 coils, each 66 inches tube length by 24 inches wide. The coils in each bank are piped for parallel cooling water flow operation. Drain troughs at each horizontal row of coil units prevent condensate from cascading over the coils below by directing the condensate to fall, out of the air stream, into the drain sump.

The cooling water source is the component cooling water system and each unit is separately connected to the cooling system supply and return headers. The automatically operated isolation valves are outside the containment. Cooling coil performance data for operation as discussed in Section 6.4.2 are presented in Table 6.4-1; data for normal operation are also shown for comparison.

6.4.3.6 Fans and Fan Motors

The circulating fans are vane axial, preset adjustable pitch, direct connected, non-overloading, single speed machines with steel casings and wheels. The cooling and filtering unit fans are 60 inch diameter and the cooling unit fans are 48 inch diameter. The fans are matched for parallel operation and designed for the air-saturated steam mixture density and flow at DBA conditions; each motor is rated for the peak of the horsepower curve at these same conditions.

The fan motors are of the totally-enclosed, air-over (TEAO) type.

The design data are presented in Table 6.4-6; data for normal operation are also shown for comparison:

Table 6.4-6 - "Fan Design and Operating Data"

Cooling and Filtering Units	Design Basis Accident Operation	Normal Operation
Item No's VA-3A & 3B		
Flow, CFM per unit	86,500	90,000
Static Pressure Rise, in. H_20	9.7	3.3
Motor HP, design operating point	215	80
Motor HP Rating per unit		227
Cooling Units, Item No's 7C & 7D		
Flow, CFM per unit	52,000	50,000
Static Pressure Rise in. H_20	8.0	3.1
Motor HP, design operating point	113	38
Motor HP Rating per unit		116

6.4.3.7 Housings, Ductwork, Exhaust Plenum and Related Accessories

The unit housings, exhaust plenum and interconnecting ductwork were constructed from reinforced galvanized carbon steel and are designed to withstand an external pressure differential of 2 psi. Relief ports (see Figure 6.4-4) are provided in the housings and plenum to open and relieve the pressure differential should it exceed 1 psi. These ports close when pressure equilibrium is restored.

The dampers downstream of the fans are of the gravity, counter-weight operated type. Flexible connections are installed at the fan duct connections to minimize the transmission of vibration.

The plenum discharge hatches are located at the bottom of the plenum and direct the discharge downwards below the operating floor. Also a single two foot diameter damper is located at the top of the plenum and directs a small portion of the discharge to the containment dome. The hatches are a series of doors normally held closed against gasketed flanges by chains with fusible links. On an increase in temperature to 160°F the links part and the counter balances open the hatches.

6.4.4 System Operation

Normal system operation is described in Section 9.10. A containment high pressure (CPHS) and/or a pressurizer low pressure signal (PPLS) (see Section 7.3.2) initiates the following:

a. VA-3A & VA-3B are started with PPLS <u>OR</u> CPHS via the sequencer.

b. VA-7C & VA-7D are started with PPLS <u>AND</u> CPHS via the sequencer.

c. On the cooling and filtering units, the face dampers open and the bypass dampers close.

d. The component cooling water system valves on the cooling coil supply and return lines receive actuation signals to open.

All four units are then operating in the emergency mode. If all normal power sources are lost and only one emergency diesel-generator functions, one cooling unit fan and one cooling and filtering unit fan operate. The operator has the capability to isolate cooling water to any air cooler with an inoperable fan to maximize cooling water flow to the operating units.

The containment atmosphere temperature increase at the commencement of the DBA melts the fusible links on the plenum hatches allowing them to spring open. Should the pressure transient across the unit housings exceed 1 psi the relief ports open automatically.

After the DBA blowdown, the containment atmosphere is a mixture of air saturated with steam at a maximum pressure and temperature of 60 psig and 288°F, respectively. This atmosphere also contains borated water droplets and mist, the bulk of which derives from the emergency containment sprays but some of which may emanate from the reactor coolant released at rupture. Water droplets and mist are removed in the moisture separators and mist eliminators and, after the particulate and adsorptive filtration (in the case of the two larger units), the air-steam mixture enters the cooling coils saturated at 288°F. Part of the water vapor condenses during the cooling process and the mixture that leaves the coils is still saturated but at a lower temperature (approximately 270°F) and higher density. The mass flow leaving the coils is less than the mass flow entering by an amount equivalent to the quantity of steam condensed; the saturated air flow leaving the coils, expressed in CFM, is therefore correspondingly less than that entering the coils. The condensate cascades into the cooling coil sump and is discharged by the drain from where it cascades into the containment sump. The air-steam mixture leaving the coils is drawn through the fans, where it absorbs some sensible heat from the fans and motors, and passes into the plenum. The cycle is completed by discharge through the plenum hatches.

An analysis was performed to determine the distribution of air in the containment after a DBA. The results of this analysis are listed below. For case I, it was assumed that the ring header, supplying air to the various regions of the containment, collapses due to the high containment pressure created by the DBA blowdown and all the air leaving the cooling units exits through the hatches at the bottom and at the top of the plenum. For case II, it was assumed that the ring header is sheared off on both sides of the air plenum and the air leaving the cooling units exit through the hatches at the bottom and at the top of the plenum as well as through the openings left by the sheared off ring header.

<u>Flow Distribution</u>

	Case I	Case II
Basement Level	20%	10%
Intermediate Level	30%	13%
Operating Level	45%	73%
Containment Dome	5%	4%

In both of the aforementioned cases, enough air is circulated through the containment to provide at least one air change per hour for each of the various regions. This rate of ventilation ensures that no hydrogen or radioactive gases will be concentrated to a dangerous level in any region. Air movement in various areas of the containment was verified with an air flow configuration representative of duct work post accident conditions after installation of the air handling system.

6.4.5 Design Evaluation

The containment air cooling and filtering system provides the design heat removal capabilities for the containment during the postulated loss-of-coolant accident. The system accomplishes this by continuously recirculating the air-steam mixture through cooling coils to transfer heat from the containment atmosphere to the component cooling water.

The system is independent of the containment spray system. These conditions would apply if one of the cooling and filtering units failed to operate at safeguards initiation. In the unlikely event that normal power sources are lost and one emergency generator fails to operate, one cooling and filtering unit and one cooling unit operate.

The performance of the cooling coils and the fans was verified by test at the DBA conditions and the pressure relief port design was also evaluated by test (see Section 1.4.8.1).

As a Class I system all components were designed to survive the seismic loadings imposed during the maximum hypothetical earthquake without damage or any change or loss of function. All components were also designed to withstand the rapid pressure increase at the commencement of the DBA. All materials and components are suitable for sustained operation at the DBA environmental pressure, temperature and humidity and are resistant to boric acid at the anticipated concentrations. The system is protected from missile damage, specifically by missile shields and inherently by the equipment locations at elevations 1045'-0"and 1060'-0" which lie outside the trajectories of any potential missiles. The coil inlet cooling water design temperature of 120°F is based on the operation of two component cooling water pumps and two raw water pumps.

The automatic, timed, sequential starting of fans and pumps and the opening of the cooling water valves (see Sections 7.3 and 8.4) is such that water and air flows are delivered within at most 60 seconds after a safeguards initiation signal.

The dampers used to direct air through the charcoal filters after the design basis accident are air piston operated and fail in the position required for post accident filtration. These pistons work against a spring so that in case of control air failure, the dampers will go to the proper position for filtering air. Also, the solenoid valve which supplies control air to the piston was designed so that electrical failure cuts off control air pressure, causing the damper to assume its failure position. Proper operation of these dampers is checked by periodic in-service tests.

The fusible link hatches have counter balances. The fusible links, which hold the hatches closed during normal operation, are in accordance with the Standard of Underwriter's Laboratory, Inc. This standard requires testing of each batch of links produced to assure that they part at the required temperature. Once the links operate, counter balances will force the hatches open. Periodic tests are performed during shutdowns to verify operability of the hatches. These tests are destructive tests and require replacement of the links.

The mixture entering the main air handling unit under DBA conditions has a density of 0.201 pounds per cubic foot. Cooling and condensation of water vapor have the effect of increasing the density to 0.215 pounds per cubic foot, an increase of 7%. The tendency will be for the cooled mixture to settle toward the bottom of the containment. Flow channels between the various levels inside the containment are provided by annular gaps between the edges of the floor and the containment shell in addition to hatchways. The aggregate area of these gaps far exceeds the cross sectional area of the duct work. Assuming the ductwork fails during the DBA, air discharged from the fans will be directed through the fusible link operated hatches, reinforcing the tendency of the cooled mixture to go to the bottom of the containment. This hatch is located over the annulus at the edge of the containment floor. If a portion of the normal ductwork remains intact, it will conduct air directly to the bottom of the containment.

The main air handling units draw air in horizontally, from the center region of the containment through inlets between elevations 1048' and 1077'. Approximately three quarters of the free volume in the containment is below elevation 1077' and will participate in the circulation system induced by density differences. Mixing above this level will be provided by the shearing effect of air being drawn horizontally across the containment by the suction of the air handling units. There are no heat sources above this level, so density differences are negligible.

The individual air handling units have no components in common with exception of a dividing wall between pairs of units and a common discharge plenum. The dividing wall is unnecessary if both units of the pair are operating. If only one unit is operating, the only effect of this wall's failure is to allow air to be drawn through the suction side components on the idle eliminators. Only in the case of cooling coils would such a rupture degrade performance, and then only if cooling water to the coils were shut off. Failure of the discharge plenum would disturb the air circulation path. The plenum has been designed to withstand the forces during the design basis accident, however, and is not considered likely to fail. The only type of failure which could result in a serious degradation of performance is a complete inwards collapse with the flow path being choked off. The plenum has been designed specifically to avoid such failure.

Failure of a damper to seal tightly will reduce the effective filtering capacity of its air handling unit by approximately the amount of leakage through the damper. It would not reduce the cooling capacity of the unit since, no matter which damper air flows through, it subsequently flows through the cooling coils.

There are 92 pressure relief valves or posts on the main air handling units. Failure of these to reseat would have the following consequences:

a. 12 are positioned such that their failure to close would allow air to bypass the cooling coils, charcoal filters, HEPA filters, and mist eliminators.

b. 8 are positioned such that their failure to close would allow air to bypass the cooling coils and mist eliminators.

c. 16 are positioned such that their failure to close would allow air to bypass the charcoal filters, HEPA filters and mist eliminators.

d. 24 are positioned such that their failure to close would allow air to bypass the HEPA filters and mist eliminators.

e. 32 are positioned such that their failure to open would allow air to bypass the mist eliminators alone.

If all 92 valves were to fail open, flow through the charcoal filters would be reduced by 11%; such an event would be extremely unlikely, however, since the relief ports are designed to close by gravity, and the bearings on these parts are designed to minimize binding. In any event, the loss in cooling capacity is less than proportional to the loss of flow through the cooling coils since heat transfer is mostly by condensation.

The number of pressure relief valves to be provided was based on limiting the pressure differential across the housing to 1.5 psi. The housing was designed to withstand 2 psi; the difference being allowance for the failure of some relief ports to open. Testing of the pressure relief ports is performed periodically by pushing them open manually.

Each of the four carbon filter banks has twelve instrumented cells among the one-hundred forty-four cells. The temperature time response of the thermistors is on the order of a very few seconds. For fires that occur during normal operation, the thermistors will alarm at 450°F. The ignition temperature of the charcoal is at least 640°F. The difference in temperature will give the operator enough time to start the charcoal dousing operation manually.

The water delivery capacity of the deluge system (design 1.5 gpm for each of the 576 nozzles) is adequate for cooling a bed with hot spots. Cooling a carbon filter which has hot spots is preferable to attempting to quench a fire which has already started. AEC-Lawrence Radiation Laboratory tests have shown that water flow rates several orders of magnitude larger than Fort Calhoun's were unable to extinguish an existing fire in a carbon bed.

6.4.6 Availability and Reliability

Since the system is operated to remove atmospheric heat loads from the containment during normal plant operation it is in a state of permanent availability to perform the emergency safeguards function in the event of an accident. At safeguards initiation the fan motor loads on the operating units are increased and idle units are started. The normal operational requirements give the advantage that the availability of the system is never an unknown quantity, but since system components are only lightly loaded during normal operation the system reliability is enhanced.

The fans are direct-connected; there are no belts or flexible couplings. No auxiliary systems are required to cool the fan motors since they are of air-over design and are cooled by the air-steam mixture circulated by the fan wheels.

The fan motors were designed for maximum operating reliability under DBA conditions. Special design objectives include adequate insulation system integrity, bearing lubrication, and internal resistance to chemical attack. The motors are also equipped with:

a. Ammeters which are located in the control room.

b. Bearings equipped with vibration detectors which initiate alarms in the control room in the event of high vibration.

Periodic testing (see Section 6.4.7) and replacement of filters ensures that these crucial components are in a state of readiness to function at their design efficiencies. Critical operating temperatures and pressure differentials are monitored during both normal and emergency operation so that any deterioration in performance can be detected. In addition, charcoal filter bed temperatures are monitored during normal operations so that fires or localized hot spots can be detected. The water side surfaces of the cooling coils can be cleaned, if required, to maintain design heat transfer capability.

The small number of active components and fail-safe design, where possible, contributes to the availability and reliability of the system. Dampers and cooling system valves are pneumatically operated and designed to assume the emergency position on loss of air pressure or electrical signal.

The operation of any one air handling unit is independent of the other units. Should a component cooling circuit leg to the coils rupture inside the containment, or a coil itself fails, a loss of return flow alarm warns the operator to isolate that circuit.

6.4.7 Tests and Inspections

The testing and inspection program ensures that the system is capable of meeting the design performance objectives and is in a constant state of availability to carry out the design function. Testing and inspection can be classified under the three categories of development tests, manufacturer's shop tests, and on-site tests.

6.4.7.1 Development Tests

The cooling coils and fan-motor units were subject to tests at the DBA pressure, temperature and humidity. The unit housing relief ports were also tested to verify their relieving capacity. These test programs are described in Section 1.4.8.1.

6.4.7.2 Shop Tests

All equipment was subjected to the test and inspection provisions of the reference codes and standards. The following information highlights some of these tests and describes some of the additional tests required by the equipment specifications.

Sample HEPA filters were subjected to air flow tests at standard conditions, a moisture resistance test, a dust loading capacity test in accordance with the NBS Standard Dust Loading Test and an ultimate strength test with the NBS dust load still on the filters. All HEPA filters were subjected to a DOP (dioctylphthalate) penetration test in accordance with MIL Standard 282. To pass this test, penetration of 0.3 micron diameter DOP particles must not exceed 0.03 percent.

Charcoal filters were subjected to air flow resistance tests in accordance with Military Specification MIL-F-50048. Each filter was tested for leakage in accordance with AEC-DP-1082.

6.4.7.3 On-Site Tests

The system was designed to facilitate the inspection, testing and replacement of all important components. Features included are access doors, access spaces between filter banks, inspection platforms and filter test injection and monitoring connections.

After installation the system was tested with regard to flow capacity and mechanical operability. These tests were carried out at standard atmospheric conditions. Dampers and the pumps and valves of associated systems were tested for operation at the proper set points. Controls, instruments and alarms were checked for operability and adequacy of limits.

The HEPA filter banks were initially in-place tested for leakage and are visually inspected for leakage during refueling outages.

Surveillance tests require flow testing for the VA-7C and VA-7D fans to satisfy the requirements of Technical Specification 3.6(3)f for the containment air cooling and filtering system. IC-ST-VA-0013 testing of the VA-7C and VA-7D fans demonstrates compliance with the mixing requirements of 4.84hr $^{-1}$ in the unsprayed region of containment. If necessary, fan blade adjustment or reanalysis will be performed to confirm the tested fan flow rates are consistent with the mixing analysis.

Facilities were provided (see Section 7.3.4) to test the full operational sequence that would bring the system (and the systems upon which it is dependent), into operation at the DBA. This includes the starting of fans and pumps, the operation of valves and dampers and the transfer to emergency power sources.