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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 1/2 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 1/2 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 $\pm$ 25°F for 1/2 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 1/2 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:

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

where $C_o = 2.3 \times 10^{-19}$ 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. The average reactor coolant temperature is programmed to vary as a function of load as shown in Figure 4.3-9. A reduction in load results in the average reactor coolant temperature dropping to its programmed value for the lower power level. 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, the average reactor coolant temperature is raised in accordance with the coolant temperature program (Figure 4.3-9). 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. A non-operating reactor coolant pump shall not be started unless at least one of the following conditions is met:

1. A pressurizer steam space of 60% by volume or greater exists, or

2. The steam generator secondary side temperature is less than 50°F above that of the reactor coolant system cold leg. Formation of a 60% steam space ensures that the resulting pressure increase would not result should a reactor coolant pump be started when the steam generator secondary side temperature is greater than that of the reactor coolant system cold leg.

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.	1/2
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. 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.

4.3.9.5 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.

4.3.10 Missile and Seismic Protection

4.3.10.1 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.

4.3.10.2 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.

4.3.11 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
Shell and Top Head	AISI 304 SS

4.3.12 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.

4.3.13 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.

4.3.14 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.

4.3.15 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.

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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 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 CMPACT 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).

5.9 CONTAINMENT PENETRATIONS

5.9.1 General

Containment penetrations were designed to withstand normal environmental conditions prevailing during plant operation and to maintain their integrity following the DBA. Exterior portions of all containment penetrations, including access openings, are located in heated enclosures in tornado protected areas.

Penetrations fall into three categories as follows:

a. Pipe penetrations, including the fuel transfer tube;

b. Electrical penetrations;

c. Access openings; the personnel air-lock and equipment access hatch.

Figures 5.9-14 and 5.9-15 locate and identify each containment penetration. For further clarity, Figure 5.9-16 locates and identifies the function of each electrical penetration.

5.9.2 Pipe Penetrations

Piping penetrating the containment was designed for pressures at least equal to the containment design pressure. In addition pipe penetrations were designed to withstand the greatest thrust the associated piping could impose under rupture conditions. Lines which penetrate the containment and contain high pressure or high temperature fluids (e.g. steam and feedwater piping) were guided to prevent the whipping associated with the fracture of a line containing high internal energy, thereby preventing excessive forces and damage to the penetration.

All penetrations including valves are located in tornado protected areas and are protected against missiles generated within the containment, thus affording additional protection of the piping between the containment and the isolation valves.

Containment isolation valves were provided in lines penetrating the containment to assure that no unrestricted release of radioactivity can occur (see Section 5.9.5).

Penetration valves and blind flanges were designed to withstand the highest temperature and highest pressure experienced during normal operation, transient conditions, and all postulated accident conditions. Protection of each line, including main steam and feedwater, necessary to preclude pipe rupture between the penetration and the first valve, was accomplished by shortening the exposed length of pipe and installing the first valve as close as possible to the containment internal or external wall dependent upon valve operating and maintenance clearances.

Pipe penetrations were of several designs depending on the temperature and pressure of the fluid being carried, and on the size of the line. The various types of penetrations are shown in Figures 5.9-1 through 5.9-10.

High temperature pipe was thermally insulated on the outside of the pipe in the air gap between the penetrating pipe and the penetration sleeve. Cooling fins were provided on the penetration sleeve to ensure that the temperature of the containment concrete is not excessive.

All penetration pipes were anchored to the containment to limit the movement of the line relative to the containment. Bellows were provided where necessary to accommodate relative movement between the pipe and the containment shell.

The basis for penetration design was the ASME Boiler and Pressure Vessel Code, Section III, Class B, and therefore, the penetration structural and leak tightness integrity is maintained. Local heating of the concrete immediately around the penetration develops compressive stresses in the concrete adjacent to the penetration, and a negligible amount of tensile stress over a large area. The mild steel reinforcing added around penetrations distributes local compressive stresses to ensure overall structural integrity.

An analysis has been performed of post-DBA thermally-induced pressurization of liquid trapped between closed containment isolation valves. The analysis concluded that pressure boundary integrity of the piping would not be comprised for penetrations susceptible to this phenomenon (Ref. 5.13.13).

5.9.3 Electrical Penetrations

Containment electrical penetrations were of the canister type furnished by the manufacturer as fully assembled, factory tested units. Field installation only requires welding the canisters into penetration pipe stubs, and attachment of cables.

5.9.3.1 Design Bases

Canister design and construction satisfy the following criteria:

a. Forty-year service life;

b. Leak testable in service from outside containment;

c. Leak rate per canister including mounting gland not above 1.5×10^{-6} standard cubic centimeters per second of helium at 69 psig;

d. Operating temperature range of 40°F to 125°F at 60 psig;

e. Capable of withstanding a temperature and pressure of 300°F and 60 psig for two hours;

5.9.3.2 Description

Canisters provide feed throughs for circuits in the following service categories; 4,160-Volt power; 600-Volt power; 600-Volt control; and instrumentation. Instrumentation includes feed throughs for thermocouple extension wire, coaxial and triaxial cable and multi-conductor cable.

Three basic feed through types meet these requirements:

a. 4,160-Volt power: A single conductor per feed through terminated at each end with a hermetically-sealed bushing;

b. 600-Volt power, control, and instrumentation: Solid, multiple conductors of required composition and configuration per feed through;

c. Coaxial and triaxial instrumentation: Circuit feed through retains coaxial and triaxial configurations for proper shielding, grounding, and impedance match; one circuit per feed through.

A piece of carbon steel pipe, called the barrel, with flat, stainless steel headers welded to each end comprise the canisters. Feed throughs pass through both headers, to which they were sealed by means of compression sealing glands. A mounting gland was welded to the barrel.

Feed throughs were fabricated of teflon-insulated, solid conductors bound in a matrix of insulating/sealing material, all held in compression within a swaged, stainless steel outer housing. Those feed through subassemblies having conductors associated with equipment required to operate post-accident have been replaced with subassemblies environmentally qualified per the EEQ program. The feed throughs that were installed are of the same design as the existing except the conductors are insulated with Kapton insulation. Two sealing sections were provided near each end of the feed through; in each such section contours place the insulating/sealing material in high compression and prevent cold flow. To prevent end-to-end gas migration along conductors, insulation was omitted from a short section of conductors just inside each pair of sealing sections and any gas appearing in these gaps is vented to the canister interior through outer housing ports. These ports also serve to transmit the canister testing/monitoring pressure to the sealing sections.

On installation, the mounting gland was field welded to the inner end of the containment penetration pipe stub, and a welding ring was welded at one edge to the canister and at the other edge to the pipe stub. A port through the barrel and mounting gland so connects the canister interior with spaces between welds provided under the mounting gland and welding rings, that pressure testing the canister includes all shop and field welds.

A second electrical penetration (type B) utilizes a single header plate on the auxiliary building side of the pipe. Support plates are located in the pipe and on the containment end. Pressure monitoring ports are provided in the header plate.

For pressure testing, a pressurizing connection with a pressure gage is provided on the outer header. This provision makes possible either periodic leak testing, or monitoring by pressurizing the canister with dry gas, sealing off, and periodically checking pressure gage indication.

Each canister was provided with an internal thermocouple for monitoring temperature.

5.9.4 Access Openings

5.9.4.1 Personnel Air Lock

The personnel air lock, shown in Figure 5.9-11, is a welded steel assembly consisting of a cylindrical barrel with attached thickened insert plate, a steel floor designed for a live load of 200 psf and a bulkhead at each end. Each bulkhead contains a 3'-6" by 6'-8" gasketed steel door. The air lock, including both doors, was designed to withstand all containment design conditions with either or both doors locked. The hatch was designed and fabricated in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class B vessels.

Each door opens inward so that any overpressure within the containment will tend to seal the door against its gaskets. The doors are mechanically interlocked to ensure that one door cannot be opened until the other door is sealed; special tools and procedures are required to override the interlock. Pressure equalizing valves and gages are provided as well as visual indication outside each door to show the status of doors and valves. An alarm annunciates in the control room if the doors of the air lock system are not completely secured. Doors are normally manually operated with backup hydraulic closure in emergencies.

An interior lighting system capable of operating from an emergency dc power supply, and an emergency communication system, are provided.

5.9.4.2 Equipment Access Hatch

The equipment access hatch, shown in Figure 5.9-12, is a welded steel assembly consisting of a 14 foot diameter cylindrical barrel with attached thickened insert plate and a dished bolted cover. Lifting attachments and guides are provided for removal of the cover, which can be unbolted only from inside the containment. The hatch was designed and fabricated in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class B vessels.

Since access to the equipment hatch for removal and/or replacement presents footing and falling hazards, a work platform has been installed so as to assure safe working conditions. The platform is 14'-10" long and varies in width from 2'-3" to 2'-9" (to conform to the radius of the containment liner). The working platform is adjacent to the equipment hatch and suspended from the above floor (elevation 1013') by four hangers from structural members B-13A. A ladder is permanently installed to provide access to this platform from the lower level of containment (elevation 994').

5.9.5 Containment Isolation System

The containment isolation system is defined as those devices actuated by a Containment Isolation Actuation Signal (CIAS) or a Steam Generator Isolation Signal (SGIS) regardless of which functioning system the device belongs. The containment isolation system is an Engineered Safety Feature System and hence, is part of the plant's Engineered Safeguards.

The containment isolation system was designed to prevent the release of radioactivity from containment, especially in the event of an accident. In the event of a Loss of Coolant Accident, the release of radioactivity is mitigated by establishing containment integrity. Containment integrity is established and maintained by automatically closing those containment isolation system valves that require closure to maintain the integrity of the containment barrier. Certain valves on Engineered Safety Feature piping systems which penetrate the containment automatically open to facilitate the accident mitigation function of their respective systems. Automatically controlled containment isolation valves are actuated by the safeguards control system as discussed in Section 7.3.

The CIAS signal to the steam generator blowdown sampling isolation valves (HCV-2506A/B and HCV-2507A/B) can be overridden in the event of a steam generator tube rupture (SGTR) that results in a CIAS. This feature aids Operations in identifying the affected steam generator. Additionally, the CIAS signal to the reactor coolant system sampling isolation valves (HCV-2504A/B), radioactive waste disposal system containment isolation valves (HCV-500A/B) and radiation monitoring containment isolation valves (PCV-742E/F/G/H) can be overridden if the requirement for sampling using the post accident sampling system (PASS) is required.

The CIAS signal to the RCS letdown stop valves can be manually overridden (Ref. 5-15) in order to reduce excessive RCS inventory during an UHE event. This situation requires adherence to the HPSI Stop and Throttle Criteria (as defined in Emergency Operating Procedures) prior to overriding CIAS from RCS letdown stop valves.

In cases where the system penetrating the containment is, or is capable of being operated during normal reactor plant operation, the penetrations are equipped with automatic or remote-manual operated valves. In a restricted number of cases, when the system is inactive during reactor operation, manually operated containment isolation valves are employed; the operation of these valves is under administrative control. Examples of such manually operated valves are those on the steam generator secondary side drain lines and certain valves in the shutdown cooling system; these latter are locked closed during reactor operation as discussed in Section 9.3.2.

The automatic isolation valves consist of air-operated valves, motor-operated valves, and solenoid-operated valves. All automatically operated containment isolation valves are designed to fail to the position that provides the greatest safety. The position of all automatic valves, open or closed, is indicated in the control room.

In certain cases check valves on influent lines penetrating the containment are classified as isolation valves. These valves close on loss of system pressure or reversal of flow.

Branch connections between the containment and the isolation valves are only included when necessary for system functioning or to enhance reliability, testing or leak tightness. Such branch lines are equipped with valves to provide isolation integrity equal to at least that of the main system.

All piping penetrating the containment and all containment isolation valves were rated for at least the containment design pressure and were designed to operate in all normal and postulated accident environments.

In all cases at least one isolation valve is located outside the containment.

Containment isolation valve arrangements are classified under three categories as follows:

a. Reactor coolant exposed systems;

b. Containment atmosphere exposed systems;

c. Closed systems.

The criteria applicable to the above system categories are defined below and the valve arrangements are shown in Figure 5.9-13.

Reactor Coolant Exposed System

Reactor Coolant Exposed Systems are those systems where valves are the boundaries between the reactor coolant and the fluid in the system.

Each main process line in a Reactor Coolant Exposed system that penetrates the containment wall should be provided with one locked closed or automatic isolation valve inside containment and one locked closed or automatic valve outside containment. (Valves on lines which branch from between the main process line isolation valves, such as drain valves, are maintained normally closed through administrative controls.) Other configurations are acceptable provided that the containment isolation provisions can be demonstrated to be acceptable on some other basis.

The automatic isolation valves inside containment are located between the containment wall and the first major component in the system, as close to the first major component as practical. The automatic isolation valves outside the containment are located as close to the containment wall as practical. Upon loss of actuating power, automatically positioned valves on systems in this category are designed to take the position that provides greater safety (i.e. isolation valves which are not required to open to support the accident-mitigating function of an ESF system are closed).

The Chemical and Volume Control System (CVCS) is classified as a Reactor Coolant Exposed system. The CVCS charging line enters containment through penetration M-3. The valve configuration on this penetration is acceptable because the pressure in the direction of flow toward containment is greater than the maximum containment pressure.

Containment Atmosphere Exposed Systems

Containment Atmosphere Exposed Systems are designated as those systems in which valves are the primary boundary between the containment atmosphere and the fluid in the system. In some cases, the system outside containment is a closed piping system.

Each line in a Containment Atmosphere Exposed system that penetrates the containment wall shall be provided with one locked closed or automatic isolation valve inside containment and one locked closed or automatic isolation valve outside containment. Other configurations are acceptable provided that the containment isolation provisions can be demonstrated to be acceptable on some other basis. A cap or blank flange on a line is equivalent to a locked closed isolation valve.

The automatic isolation valves outside the containment are located as close to the containment wall as practical. Upon loss of actuating power, automatically positioned valves on systems in this category are designed to take the position that provides greater safety (i.e. isolation valves which are not required to open to support the accident-mitigating function of an ESF system are closed).

The definition of a Containment Atmosphere Exposed System as used in this section is not the same as that used in the General Design Criteria. The Demineralized Water, Service Air, and Instrument Air penetrations are classified in this section as Containment Atmosphere Exposed systems, but would be classified as closed systems under the General Design Criteria.

Penetrations M-38, M-50, M-51, and M-52 are associated with safety related containment pressure instrumentation. Each penetration contains one remote-operated valve outside containment. Open/Closed position indication of these valves is provided in the control room. Since the associated containment pressure instrumentation is intended to remain in service after an accident, the valves on these penetrations are required to remain open. Since the valves remain open post-accident and these penetrations are associated with closed systems outside containment, the single isolation valve design is acceptable.

The Containment Sump ECCS Recirculation penetrations (M-HCV-383-3 and M-HCV-383-4) each contain one motor-operated isolation valve outside containment. Isolation of these penetrations is enhanced by the enclosure of each penetration's isolation valve, including the piping between the containment and the valve, in a protective housing to prevent leakage to the Auxiliary Building atmosphere (refer to USAR Section 6.2.3.8). The valves on these penetrations automatically open to initiate the recirculation phase of a LOCA. The single isolation valve design of these penetrations is acceptable because of their post-accident operating mode and because they are associated with a closed system outside containment.

Containment Spray (CS) is classified as a containment atmosphere exposed system. The CS headers enter containment through penetrations M-86 and M-89. The valve arrangement is shown in Figure 5.9-13 for these penetrations. In the event of a LOCA, the CS system will operate. When the system is operating, the pressure seen in the direction of flow toward containment is greater than the maximum allowable containment pressure Pa (60 psig). The times when the CS system is not operating post-LOCA are (1) during the short delay time (due to sequencer settings) prior to starting of the CS pumps, and (2) when containment spray is terminated later in the accident. It has been verified through analysis that a containment atmospheric leakage path is not created via the CS lines during the time delay prior to CS pump starting. Emergency Operating Procedures allow termination of CS flow if containment pressure is less than 3 psig. This pressure is not sufficient to overcome the liquid loop seal configuration remaining in the CS piping after termination of containment spray, so containment atmospheric leakage is precluded.

Penetrations M-30 and M-69 are associated with the Hydrogen Purge system, which would operate in a post-accident condition. Each penetration contains an automatic air-operated valve inside containment and a locked-closed manual valve outside containment. The valves inside containment are closed by a CIAS. The valves inside containment fail open on a loss of control power or instrument air, however a loss of instrument air will not create a loss of containment integrity on these penetrations because backup containment isolation is provided by the redundant locked closed manual valves outside containment.

Penetrations M-57 and M-58 are associated with hydrogen analyzer VA-81B, which would operate in a post-accident condition. Each penetration contains an automatic air-operated valve inside containment and an automatic solenoid-operated valve outside containment which are closed by a CIAS. The valves inside containment fail open on a loss of control power or instrument air, and the valves outside containment fail closed on a loss of control power. A loss of instrument air will not create a loss of containment integrity because (1) backup containment isolation is provided by the redundant fail-closed valves outside containment, and (2) the hydrogen analyzers are a closed system outside containment and are intended to operate in a post-accident condition.

The Demineralized Water Supply line containment penetration (M-80) contains two automatic isolation valves outside containment. These valves are normally closed, as this is a service system used primarily during refueling outage periods. The containment sump to Waste Disposal (WD) system (penetration M-8) contains two automatic isolation valves outside containment. These valves are normally open to facilitate the discharge of the containment sump pumps to the WD system for processing. Since the containment has a limited access, the isolation valves of penetrations M-80 and M-8 can be better inspected and maintained from outside the containment.

Closed Systems

Closed Systems are designated as those systems that are not reactor coolant exposed systems, where the fluid is separated from the containment atmosphere by a continuous barrier. The continuous barrier normally consists of a closed piping system inside containment, which may contain some small vent, drain, or relief valves.

Except where described below, lines in Closed Systems that penetrate the containment wall are provided with two isolation valves. As a minimum, one of these valves is located outside containment, and may be either automatic, locked closed, or capable of remote-manual operation.

Except where described below, those penetrations associated with non-safety related Closed Systems are provided with two automatic, fail-closed isolation valves.

Isolation valve(s) outside the containment are located as close to the containment wall as practical. Upon loss of actuating power, automatically positioned or remote-manual valves on systems in this category are designed to take the position that provides greater safety (i.e. isolation valves which are not required to open to support the accident-mitigating function of an ESF system are closed).

Penetrations M-94 and M-95 are the Main Steam lines penetrating the containment. Each penetration is provided with a single Main Steam Isolation Valve (MSIV) outside of containment. Inside containment, the system is exposed to the secondary side of the steam generators. A fail-closed containment isolation valve inside containment is not desirable, because it would isolate the secondary safety relief valves and potentially cause overpressurization of the secondary side of the steam generator and associated equipment. The single Main Steam Isolation Valves are also backed up by other isolation valves. Utilization of non-safety grade components as a backup to the MSIVs is acceptable because the design and performance of the non-safety components are compatible with the accident conditions in which they might be called upon to function. The single MSIV on each of these penetrations is therefore acceptable.

The auxiliary feedwater system provides the primary means for removal of decay heat from the steam generators in the event of a loss of electrical power. It serves the same function following a main feedwater line break. It is, therefore, essential that an open path to the steam generators is available under the above conditions. An alternate flow path for auxiliary feedwater is available using the main feedwater system.

Since the main feedwater is a closed system with check valves inside the containment, the installation of "fail-open" valves outside of the containment will not provide any additional isolation protection. The possibility of interruption of the feedwater flow to the steam generators, as a result of an air failure, makes "fail-closed" isolation valves unsuitable. Therefore, motor operated "fail as is" valves were chosen for the main feedwater isolation.

The main steam system supplies steam from both main steam lines to the auxiliary feedwater pump FW-10. The isolation valves for the lines that supply FW-10 (YCV-1045A and YCV-1045B) fail open to ensure there is a steam supply to FW-10. These valves have air accumulators to ensure that they can be closed in the event of a steam generator tube rupture.

The containment isolation valves for supply and return of component cooling water to the reactor coolant pump seal coolers and CEDM seal coolers remain open upon receipt of a CIAS signal. If power is available, the reactor coolant pumps can continue to operate and assist in providing boric acid to the core. The containment isolation valves HCV-438A, B, C, and D close on receipt of a CIAS and a component cooling water pressure low signal present for 30 seconds or more. This will ensure that under accident conditions when the CCW system becomes depressurized, containment isolation can be maintained.

5.9.6 Testing

5.9.6.1 Pipe Penetrations and Isolation Valves

All penetrations and isolation valves were tested in accordance with the applicable codes and standards. Additional tests were performed on certain items. For example, the 42 inch diameter containment purge system isolation valves were tested with 300°F air at 60 psig. The standard of acceptability was a leak rate no greater than 0.01 SCFH at the 60 psi pressure differential.

All pipe penetrations are provided with test connections. The penetrations can be leak tested without pressurizing the containment.

The safeguards automatic control system actuates the isolation valves under accident conditions. As discussed in Section 7.3.4, safeguards controls and safeguards components are periodically tested as systems insofar as is feasible. Isolation valves cannot be included in such periodic tests with the plant operating at power, however, without inducing plant trip and risking equipment damage. It is necessary, therefore, to test and exercise isolation valves individually when operating conditions permit; such isolation valves as those in the main steam and instrument air lines are exceptions to this type of testing with the plant at power. Overall isolation valve testing on a system basis may be carried out when the plant is in shutdown.

5.9.6.2 Electrical Penetrations

Electrical Penetrations canisters are subject to tests in four categories:

a. Manufacturer's shop tests;

b. Type qualification tests and analysis under simulated accident environment conditions;

c. Leak tests following installation in the containment;

d. Periodic inservice leak tests.

Periodic leak tests are performed. Such tests are performed from outside the containment, testing canisters individually. A pressurizing port and gage is provided for each canister. As arranged, the applied pressure difference exists across every seal and weld. More sensitive tests with a tracer gas and sniffer can be made as necessary when the plant is shutdown and access to the inside ends of penetrations is possible.

5.9.6.3 Access Openings

Access openings were tested during the overall containment leak test (see Section 5.10.5). These openings were also tested individually.

Double gaskets were provided on each personnel air lock door to permit pressurizing to demonstrate leak tightness without interfering with the normal operation of the plant. Auxiliary restraint beams are attached to the inner door in this case to resist the pressure within the lock chamber. The equipment access hatch has also been provided with double gaskets to permit testing in a similar manner to the personnel air lock.

Table 5.9-1 - "Containment Penetration Isolation Valves"

This table has been superseded by Figure 5.9-13 Shts 1 to 66.

5.12 Specific References

5-1. Diagonal Tension Cracking in Concrete Beams Subject to Axial Forces, Mattock, A. H., paper presented to the American Society of Civil Engineers Structural Engineering Conference, Miami, February 1966.

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5-4. A Study of Stress Relaxation in Prestressing Reinforcement, Magura, D. D., Sozen, M. A., and Siess, C. P., PCI Journal, April 1964.

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5-9. Report No. 324, Appendix B, Department of Structural Engineering, School of Civil Engineering, Cornell University.

5-10. State of Stress in a Circular Cylindrical Shell with a Circular Hole, Eringen, A. C., Naghdi, A. K., and Thiel, C. C., Bulletin No. 102 of the Welding Research Council.

5-11. Local Stresses in Spherical and Cylindrical Shells due to External Loadings, Wichman, K. R., Hopper, A. G., and Meshon, J. L., Bulletin No. 107 of the Welding Research Council.

5-12. Design of Liners for Reactor Vessels, Young and Tate, Submitted to the Conference on Prestressed Concrete Vessels, London, March, 1967.

5-13. Research on Determining the Density of Sands by Spoon Penetrating Testing, Gibbs, H. J., and Holtz, W. G., Earth Laboratory Branch, Bureau of Reclamation, presented to the 4th International Conference on Soil Mechanics, 1957.

5-14. EA-FC-96-041, Auxiliary Building Tornado Venting Analysis

5-15. NRC Safety Evaluation Report related to Amendment 191 concerning Unreviewed Safety Question (USQ) on overriding the containment isolation actuation signal closure signal, July 22, 1999.

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

6.2 SAFETY INJECTION SYSTEM

6.2.1 Design Bases

The system is designed to prevent fuel and cladding damage that could interfere with adequate emergency core cooling, and to limit the cladding-water reaction to less than approximately 1 percent for all break sizes in the primary system piping up to and including the double-ended rupture of the largest reactor coolant pipe, for any break location, and for the applicable break time.

The safety injection system also provides rapid injection of large quantities of borated water for added shutdown capability during rapid cooldown of the reactor coolant system caused by a rupture of a main steam line. No fuel damage would result from this accident with safety injection system operation, even with the most reactive control element assembly (CEA) stuck in its fully withdrawn position. The system requirements during a main steam line rupture are discussed in Section 14.12, "Main Steam Line Break Accident."

The system requirements during a Design Basis Large Break LOCA are met with the assumption of three of the four safety injection tanks delivering borated water to the core and with one high pressure injection pump delivering approximately 75 percent of its rated flow to the core and one low-pressure injection pump delivering approximately 75 percent of its rated flow to the core.

Part of the system is also used to remove heat from the reactor coolant system for normal cooldown and to maintain a suitable temperature for refueling and maintenance as discussed in Section 9.3.

The system is designed to keep the core covered for extended periods of time after initial injection. One high-pressure pump has sufficient capacity with 25 percent spillage to maintain the core water level at the start of recirculation, and during Long Term Core Cooling (LTCC). However, to mitigate control problems that may be caused by hardware limitations, procedures require that two HPSI pumps be utilized during simultaneous hot and cold leg injection. If only one HPSI pump is operable, then use of the alternate hot leg injection path and one LPSI pump is recommended.

In the shutdown cooling mode, the system was designed to cool the reactor coolant system from 300°F to 130°F (see Section 9.3).

The system was designed to the Class 1 seismic criteria as delineated in Appendix F. The system was designed to withstand the appropriate seismic load simultaneously with other applicable loads without loss of function.

Adequate NPSH is available in the redundant ECCS injection and containment spray systems as shown and explained in the calculations given below.

I. Safety Injection

During the safety injection operating mode, the HPSI, LPSI, and the containment spray pumps receive suction from the safety injection refueling water tank. The available NPSH for the HPSI, LPSI and containment spray pumps during safety injection was evaluated as follows:

$$\text{NPSH (available)} = \frac{(P+Pa-Pvp)}{\text{sp.gravity}}\, 2.31 + Z - h_f$$

P = pressure on liquid in SIRW tank or sump

Pa = atmospheric pressure (psia)

Pvp = vapor pressure of liquid at specific temperature T (psia)

Z = vertical distance from liquid surface to centerline of pump suction (ft)

h = water level above baseline elevation of SIRWT (ft)

h_f = max. friction losses in suction lines (ft)

A. LPSI Pumps

Required NPSH of LPSI pumps (2940 gpm/pump)= 28 ft.

Centerline suction nozzle elevation	= 973.25 ft.
Minimum water level elevation in SIRWT	= 989 ft.
P (psig)	= 0.0
Pa (psia)	= 14.2
Pvp @ 105°F (psia)	= 1.10
Z (ft)	= 15.75
h_f maximum (ft)	= 7.03

Minimum NPSH (available) = $\frac{(0+14.2-1.10)}{1.0}$ (2.31)+(15.75-7.03)= 38.98 ft.

The available NPSH of 38.98 feet during the injection phase is above the required NPSH of 28 ft at a LPSI pump flow rate of 2940 gpm.

B. Containment Spray Pumps

Required NPSH of containment spray pumps (3100 gpm/pump) = 27.3 ft.

Centerline suction nozzle elevation in SIRWT	= 973.25 ft.
Minimum water level elevation in SIRWT	= 989 ft.
P (psig)	= 0.0
Pa (psia)	= 14.2
Pvp @ 105°F (psia)	= 1.10
Z (ft)	= 15.75 ft.
h_f maximum (ft)	= 9.32

Minimum NPSH (available) = $\frac{(0+14.2-1.10)}{1.0}$ (2.31)+(15.75-9.32)= 36.69 ft.

Since the minimum available NPSH of 36.69 ft. is greater than the required NPSH of 27.3 ft., adequate NPSH will always be available to the containment spray pumps during the safety injection operating mode.

C. HPSI Pumps

Required NPSH of HPSI pumps (495 gpm/pump)	=	17.6 ft.
HPSI suction nozzle elevation ft.	=	972.67
Minimum water level elevation in SIRWT	=	989 ft.
P (psig)	=	0.0
Pa (psia)	=	14.2
Pvp @ 105°F (psia)	=	1.10
Z (ft)	=	16.33
h_f maximum (ft)	=	13.02

$$\text{Minimum NPSH (available)} = \frac{(0+14.2-1.10)}{1.0}(2.31)+(16.33-13.02) = 33.57 \text{ ft.}$$

Since the maximum available NPSH of 33.57 ft. is above the required NPSH of 17.6 ft., adequate NPSH is always available to the HPSI pumps during the safety injection operating mode.

Summarizing the above NPSH evaluations for the HPSI and containment spray pumps, it is concluded that adequate NPSH will be available for these pumps during the injection phase.

II. Recirculation

A. Containment Spray Pumps

Required NPSH of containment spray pumps (3100 gpm/pump)		= 27.3 ft.
True centerline suction elevation		= 973.25 ft.
Minimum water level elevation inside containment		= 996.8 ft.
Subcooling	P (ft)	= 8.99*
	Z (ft)	= 23.55
	h_f maximum (ft)	= 4.14

Minimum NPSH (available) = (23.55+8.99-4.14) = 28.4 ft.

Since the suction pressure NPSH is above the required NPSH of the CS pumps, adequate NPSH will always be available during recirculation.

B. LPSI Pumps

Note that the LPSI pumps are not required for recirculation, and they will be secured upon receipt of the recirculation actuation signal.

C. HPSI Pumps

NPSH required for HPSI pumps (450 gpm/pump)		= 13.9 ft.
Centerline suction nozzle elevation		= 972.67 ft.
Minimum water level elevation inside containment		= 996.8 ft.
Subcooling	P (ft)	= 8.99*
	Z (ft)	= 24.13 ft.
	h_f maximum (ft)	= 6.11

Minimum NPSH (available) = (24.13+8.99-6.11) = 27.01 ft.

Since the suction pressure NPSH is above the required NPSH of the HPSI pumps, adequate NPSH will always be available during recirculation.

A comparison between the minimum available NPSH and the required NPSH of the containment spray pumps and HPSI pumps leads to the conclusion that adequate NPSH will be always available to the above mentioned pumps during the recirculation mode of operation.

* - Credit is taken for subcooling. The subcooling head is above the vapor pressure of the water; therefore, vapor pressure is appropriately considered.

Figures 6.2-1 and 6.2-2 show pump elevations and piping runs.

A portion of the recirculation piping shown in Figure 6.2-1 is buried directly in concrete. Under post accident conditions compressive thermal stresses will occur in the pipe. These thermal stresses will not cause failure of the piping since the stainless steel is a ductile material and the stresses are compressive. Reinforcing bars will absorb the tensile stress in the surrounding concrete.

If degradation of the buried piping is suspected, it can be inspected from the inside.

6.2.2 System Description

The safety injection system and the containment spray system piping and instrument diagram is shown in P&ID E-23866-210-130.

The safety injection system for this plant consists of both passive and active components. The four pressurized safety injection tanks are of the passive type and require no outside power or safety injection actuation signal to operate. The safety injection tanks inject large quantities of borated water into the reactor coolant system following a large pipe break. The water rapidly covers and cools the core, thereby limiting clad melting and metal water reaction. The separate and independent tanks are each connected to one of the four safety injection nozzles; one nozzle is located on each of the four reactor coolant system cold legs. The driving head for water injection is provided by a nitrogen cover gas at a pressure of 240 psig minimum. As the reactor coolant system pressure falls below tank pressure, check valves open in the line connecting each tank to the system. Thus, these tanks will initiate their discharge when the reactor coolant pressure drops below approximately 240 psig minimum.

The active components which require safeguard actuation signals include the high and low pressure safety injection pumps as well as the containment spray pumps.

Safety injection is initiated by either a low-pressure signal from the pressurizer or a high containment pressure signal. A description of the derivation of the safeguard actuation signals is presented in Section 7.3.2.

The safeguard actuation signals start the three high-pressure and the two low-pressure safety injection pumps via the sequencers and also open the twelve safety injection line isolation valves. The safety injection pumps take suction from two independent suction headers which are supplied with borated water from the safety injection and refueling water tank (SIRWT). The pumps discharge into the reactor coolant system through the four safety injection nozzles.

For long term core cooling, a continuous source of borated water is provided by recirculating containment water. Recirculation is automatically initiated by low water level in the SIRWT; transfer to the recirculation mode may also be manually initiated. The automatic recirculation signal shuts down the low-pressure safety injection pumps, opens both recirculation line isolation valves, closes the two SIRWT suction header isolation valves and both minimum flow line isolation valves to isolate the tank from the recirculated containment water. The high-pressure safety injection pumps continue to operate in order to provide core cooling water for the complete spectrum of break sizes.

In the recirculating mode, the high-pressure safety injection pumps take suction directly from the containment. At the discretion of the operator, a portion of the cooled water from the containment spray system may be diverted to the suction of the high-pressure injection pumps. This is a preferred method of operation, but is not necessary to meet core cooling requirements. The low-pressure safety injection pumps may be used to inject cooled water when the system pressure permits.

For the post-LOCA long term cooling, a provision is made for maintaining core cooling and boric acid flushing by simultaneous hot and cold leg injection. The safety injection flow is injected simultaneously into hot and cold legs through the pressurizer auxiliary spray system and safety injection nozzles, respectively. This injection mode provides cooling for the reactor coolant system and prevents boric acid precipitation/accumulation in the reactor vessel following a LOCA.

The safety injection system also functions to inject large quantities of borated water to provide additional negative reactivity during the rapid cooldown of the reactor coolant system caused by a main steam line break.

6.2.3 System Components

6.2.3.1 Safety Injection and Refueling Water Tank (SIRWT)

The SIRWT contains a minimum of 283,000 gallons (minimum tank volume is 311,000 gallons at 180") of water above 16" recirculation level at or above the refueling boron concentration. (The USAR Chapter 14 loss of coolant accident analysis assumes a volume of 250,000 gallons of water is pumped from the SIRWT when recirculation begins. A minimum useable volume of 283,000 gallons, as required by the Technical Specifications, provides sufficient inventory to account for instrument uncertainty.) This is sufficient boron concentration to provide a 5 percent shutdown margin in a new core with all CEA's withdrawn and at a temperature of 68°F. This is sufficient water to fill the refueling cavity in containment. An additional 45,000 - 50,000 gallons is needed to fill the fuel transfer canal in the Auxiliary Building. With all injection pumps and containment spray pumps running, the tank will provide at least a 20 minute supply of water before the pump suction must be switched to the containment recirculation line inlet at elevation 994'-0". The tank is equipped with two separate full capacity outlets and three 4-inch vents which are sufficient to permit the tank to empty during an accident mode without significantly reducing the NPSH required for the safety injection pumps. The concrete tank was designed and constructed in accordance with ACI 318-63, and is lined with carbon steel plate. The liner is coated with a water resistant paint suitable for continuous immersion.

6.2.3.2 Low-Pressure Safety Injection Pumps

The low-pressure safety injection pumps are used to inject large quantities of borated water into the reactor coolant system. They are also used to circulate reactor coolant during shutdown to remove residual and decay heat. There are two pumps, either of which can circulate sufficient water to keep the temperature rise through the core to less than the full power value with the reactor shutdown at the end of core life. The pumps were designed for thermal transients which exceed those expected in operation.

Testing included a thermal transient test on one unit under the following conditions:

a. Suction temperature increase from 50° F to 303° F in 12 seconds.

b. Suction temperature decrease from 308° F to 60° F in 12 seconds.

The pumps are of the horizontal, single stage centrifugal type and are provided with mechanical face seals backed up by a bushing, with a leak-off to collect the leakage past the seal. The seals are designed for operation at 300°F. To prolong seal life, a portion of the pump discharge is cooled by component cooling water and is used to cool the seals. In addition, normal cooling by the component cooling water system is backed up by cooling water from the raw water line serving the engineered safeguards equipment (see Sections 9.7 and 9.8). Further, a low flow alarm is provided on the seal cooling water to the pumps. 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. The pump motor is capable of starting and accelerating the pump to full speed with 70 percent of rated voltage. The pumps are provided with drain and flushing connections to permit reduction of radiation levels before maintenance. The pressure-containing parts were fabricated from stainless steel; the internals were selected for compatibility with boric acid. The pumps are provided with minimum flow protection to prevent damage when starting against a closed system. The low-pressure pump data are summarized in Table 6.2-1.

Table 6.2-1 - "Low-pressure Safety Injection Pump Data"

Item No's.	SI-1A & 1B
Quantity	2
Type	Single Stage, Horizontal, Centrifugal
Motor Nameplate Voltage	4160
Design Pressure, psig	500
Design Temperature, °F	350
Design Flow Rate (per pump), gpm	1500
Design Head, ft	403
Pumped Fluid	Borated Water
Temperature of Pumped Fluid, °F	50-300 **
Shutoff Head, ft	450
Maximum Flow (per pump), gpm	2400
Head at Maximum Flow, ft	363
Basic Material	316 SS
Horsepower	300
Shaft Seals	Mechanical
Starting Time, (at 70% voltage) seconds	8
Starting Time, (at rated voltage) seconds	4
Minimum Flow (per pump), (recirculation) gpm	200*
NPSH Required at 1500 gpm, ft	14.5
Design Maximum Suction Pressure, psig	300

*Each pump's minimum flow orifice was designed to pass this flow at pump shutoff head with a given pump running individually. Pump flow will be below this value with multiple pumps running in the minimum recirculation mode. However, the actual flows achieved by the pumps in the simultaneous minimum recirculation mode have been proven to be adequate to preclude pump difficulty for the limited time the pumps could operate in that mode during an accident.

**An operability evaluation (Reference 6.5-6) performed for the LPSI pumps concluded the pumps were qualified for thermal transients from 50°F to 300°F.

6.2.3.3 High-Pressure Safety Injection Pumps

The high-pressure safety injection pumps inject borated water at high pressure into the reactor coolant system during emergency conditions. The pumps were sized to ensure that following the rapid depressurization of the reactor coolant system and re-covering of the core by the safety injection tanks, one high-pressure pump will keep the core covered at the start of recirculation, assuming 25 percent spillage. At 30 minutes into a large break LOCA there is sufficient HPSI flow to remove decay heat and keep the core covered with 35% spillage. The requirements for boron injection for the steam line break and the injection requirements for smaller break sizes were also considered in the sizing. The pumps were designed for thermal transients which exceed those expected in operation.

Testing included a thermal transient test on one unit under the following conditions:

a. Suction temperature increase from 85° F to 308° F in 5 seconds.

b. Suction temperature decrease from 323° F to 70° F in 7 seconds.

The pumps are ten-stage horizontal centrifugal units. Mechanical seals are used and are provided with leak-offs to collect any leakage past the seals. The seals were designed for operation in excess of 300°F, but are provided with cooling to extend seal life in a manner similar to the low-pressure pumps; the seal backup cooling is also similar to that provided for the low-pressure pumps. 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. The pump motor is capable of starting and accelerating the pump to full speed with 70 percent of rated voltage. The pumps are provided with drain and flushing connections to permit reduction of the radiation levels before maintenance. The pressure containing parts of the pump are stainless steel with internals selected for compatibility with boric acid. The materials were analyzed to ensure that differential expansion during the design transients can be accommodated with the clearances selected. The pumps are provided with minimum flow protection to prevent damage resulting from operation against a closed discharge.

A full scale hydraulic test was performed on each pump assembly.

All pump test setups, test procedures and instrumentation were in accordance with the Standards of the Hydraulic Institute and the ASME Power Test Code, PTC-8.2. This included verification of satisfactory operation at the stated NPSH. It also included measurement of motor starting and operating current during thermal transient testing.

The high-pressure pump data summary is shown in Table 6.2-2.

Table 6.2-2 - High-pressure Safety Injection Pump Data"

Item No's.	SI-2A & 2B & 2C
Quantity	3
Type	Ten-stage, Horizontal, Centrifugal
Motor Nameplate Voltage	460
Design Pressure, psig	1735
Design Temperature, °F	300
Design Flow Rate (per pump), gpm	150
Design Head, ft	2800
Pumped Fluid	Borated Water
Temperature of Pumped Fluid, °F	50-300 **
Shutoff Head, ft	3180
Maximum Flow (per pump), gpm	400
Head at Maximum Flow, ft	1200
Material	Stainless Steel
Horsepower	300
Shaft Seals	Mechanical
Acceleration Time (at rated voltage), seconds	4
Minimum Flow (per pump), gpm	35*
NPSH Required at 400 gpm, ft	11
Design Maximum Suction Pressure, psig	210

* Each pump's minimum flow orifice was designed to pass this flow at pump shutoff head with a given pump running individually. Pump flow will be below this value with multiple pumps running in the minimum recirculation mode. However, the actual flows achieved by the pumps in the simultaneous minimum recirculation mode have been proven to be adequate to preclude pump difficulty for the limited time the pumps could operate in that mode during an accident.

**An operability evaluation (Reference 6.5-6) performed for the HPSI pumps concluded the pumps were qualified for thermal transients from 50° F to 300° F.

6.2.3.4 Shutdown Cooling Heat Exchangers

The shutdown cooling heat exchangers are used to remove decay and sensible heat during plant cooldowns and cold shutdowns. The units were sized to remove 37,100,000 BTU/hr, based on 4500 gpm of cooling water at 93°F inlet temperature and 3000 gpm of reactor coolant at 140°F inlet temperature. The units were further specified to accept a 70°F to 300°F transient when the containment spray pump suction is switched to the containment recirculation line inlet at elevation 994'-0". The units were designed and constructed to the standards of ASME, Section III, Classes A and C, and TEMA Class R requirements. In addition to the requirements of the code, a fatigue analysis was performed which considered all specified transient conditions. The units are of a U-tube design with two tube side passes and a single shell side pass. The tubes are austenitic stainless steel and the shell is carbon steel. The data summary for the shutdown heat exchangers is given in Table 6.2-3.

Table 6.2-3 - "Shutdown Cooling Heat Exchanger Data"

Item No's.	AC-4A & 4B
Quantity	2
Type	Shell and Tube
Codes	
Tube Side	ASME Section III, Class A
Shell Side	ASME Section III, Class C
Tube Side	
Fluid	Reactor Coolant
Design Pressure, psig	500
Design Temperature, °F	350
Pressure Loss, (@3000 gpm), psi	6.0
Materials	Austenitic Stainless Steel
Shell Side	
Fluid	Component Cooling Water
Design Pressure, psig	150
Design Temperature, °F	300
Pressure Loss, (@4500 gpm), psi	5.3
Materials	Carbon Steel

Classification of the shutdown heat exchanger tube side was changed to ASME Section III, Class C. The following reasons are considered sufficient justification for acceptance of this classification:

a. The shutdown heat exchangers are normally isolated from the reactor coolant system with valves which are remotely operated from the control room. There are a minimum of two such valves in the flow path between the heat exchangers and the reactor coolant system, in addition to locally operated valves.

b. There are two redundant heat exchangers.

c. In addition to ASME Section III, Class C requirements, all the design, fabrication, inspection, and traceability requirements for Class A vessels have been met.

6.2.3.5 Safety Injection Tanks

The four safety injection tanks are used to flood the core with borated water following a depressurization of the reactor coolant system. The tanks are sized to ensure that three of the four tanks will provide sufficient water to re-cover the core following a DBA. The tanks contain borated water at or above the refueling boron concentration and are pressurized with nitrogen at 240 psig minimum.

Level and pressure instrumentation is provided to monitor the availability of the tanks during plant operation. Provisions have been made for sampling, filling, draining, relieving, venting and correcting the boron concentration. The tanks are carbon steel internally clad with stainless steel. Design, construction and overpressure protection were in accordance with the ASME Code Section III, Class C. The vessel was altered to upgrade the pressure from 250 psi to 275 psi and the supplemental data report to meet the requirements of Section VIII of the ASME code has been prepared. The data summary for the safety injection tanks is given in Table 6.2-4.

Table 6.2-4 - "Safety Injection Tank Design Data"

Item No's.	SI-6A, 6B, 6C & 6D
Quantity	4
Volume, ft^3	1300
Pressure, psig	275
Temperature, °F	200

6.2.3.6 Safety Injection Valves

The safety injection valves are electric motor operated globe valves located inside the containment on each of the high pressure and low pressure injection lines. These valves are closed during normal plant operation and were designed for the reactor coolant system pressure and temperature. The motor operators automatically drive the valves to the full open position in less than 12 seconds following the SIAS. Design data for the high and low pressure injection valves are summarized in Table 6.2. The testing and qualification of the valves at DBA conditions is discussed in Section 1.6.

Table 6.2-5 - "Safety Injection Valve Data"

	Low Pressure	High Pressure
Item No's. HCV-	327, 329, 331 & 333	311, 312, 314, 315, 317, 318, 320 & 321
Quantity	4	8
Type	Globe	Globe
Size, inches	4	2
Design Pressure, psig	2485	2735
Design Temperature, °F	650	650
Full Open Design Conditions		
C_y	130	13
Flow, gpm	1000	225
Pressure Drop, psi	55	270
Body Material	ASTM A-35, Gr. CF8	316 SS

6.2.3.7 Relief Valves

Thermal relief valves are provided in normally isolated piping to relieve a flow rate equivalent to the expansion of water due to a sudden increase in fluid temperature.

The relief valves on the shutdown cooling piping outside the containment have sufficient capacity for the operation of one charging pump with the reactor coolant system solid and not exceeding shutdown cooling piping design pressure.

The safety injection test and leakage line is provided with relief valve protection. The valve is sized to pass 40 gpm.

6.2.3.8 Piping

The safety injection system piping is austenitic stainless steel. It conforms with the standards set forth in USAS B31.7, with the exceptions of the recirculation piping from the containment sump to the containment isolation valves, and the piping from the SIRWT to the tank isolation valves; this piping is in accordance with USAS B31.1. Thermal flexibility and seismic loading analyses have confirmed the adequacy of the system piping. The design criteria are presented in Appendix F.

Safety Injection Piping

The following tests were performed to ensure the quality of fabrication and erection:

a. Piping shop welds were 100 percent radiographically inspected;

b. Piping was hydrostatically tested at the mill in accordance with the appropriate ASTM specifications;

c. Cast valve bodies were 100 percent radiographically and dye penetrant inspected;

d. Valve bodies were hydrostatically tested in the shop in accordance with ASTM specifications;

e. Field welds were 100 percent radiographically inspected;

f. Field welds were hydrostatically tested in accordance with USAS B31.7 with the exception of the piping from the SIRWT to the tank isolation valves.

Recirculation Piping

Engineered safeguards piping for containment recirculation connected to the containment is an extension of reactor containment during the recirculation mode of core and containment cooling. The following details pertain to suction piping from the sump to the first isolation valve:

a. The piping has a nominal wall thickness of 0.375 inch which results in a maximum allowable pressure for the pipe minimum wall thickness of at least 8 times the maximum expected pressure of 55 psig;

b. All shop piping welds were 100 percent radiographically inspected;

c. The piping was hydrostatically tested in the shop at 705 psig;

d. The isolation valves are rated at 150 pounds USASI based on flange ratings. The valves can withstand 210 psig at 300°F;

e. The valve bodies were 100 percent radiographically and dye penetrant inspected;

f. The field welds were hydrostatically tested in accordance with USAS B31.7, with the exception of the piping from the containment recirculation inlet to the containment isolation valves;

g. The two isolation valves in the recirculation lines and the piping between these valves and the containment penetrations are contained within pressure vessels. Each vessel is embedded in the concrete at one end and joined to the pipe, downstream of the valve, by a bellows assembly at the other. In the event of leakage, either at the valve or between the valve and the containment, the vessel maintains the integrity of the containment boundary under DBA conditions. The vessels were designed and fabricated in accordance with the requirements for Class B vessels of Section III of the ASME Boiler and Pressure Vessel Code.

6.2.3.9 Instrumentation

Temperature

The discharge temperature of each shutdown heat exchanger can be monitored in the control room.

The temperature of water extracted from the reactor coolant system during cooldown and the temperature of low pressure safety injection water returned to the reactor coolant loop is indicated in the control room.

Pressure

Pressure in each safety injection header is indicated in the control room.

Pressure between the injection check valves is indicated in the control room. Pressure is individually controlled in each of the four injection lines.

The pressure of each safety injection tank is indicated in the control room. Redundant high-and-low-pressure alarms are provided.

Level

The water level in the safety injection and refueling water tank is monitored by either of two separate level indicators in the control room. Each indicator is equipped with a low-level and a high-level alarm.

Level instrumentation mounted on each safety injection tank provides indication in the control room. Redundant high-and-low-level alarms on each tank are provided.

Containment sump water level indication is provided by redundant indicators in the control room with high-and-low-level alarms. Additional alarms have been installed to indicate containment water level in case of an accident.

Water level in each engineered safeguards pump room sump and the recirculation line isolation valve room sump is indicated in the control room via the ERF Computer.

Flow

The total low-pressure injection flow rate is measured by an orifice meter installed in the low-pressure injection header and is indicated in the control room. Each of the four low-pressure injection branch lines and each of the four high-pressure branch lines are equipped with flowmeters which can be used to balance injection flow rates.

A flowmeter installed in the safety injection test and leakage return line is used during operational tests of the safety injection system.

6.2.4 System Operation

6.2.4.1 General

Borated water is injected into the reactor coolant system by the safety injection tanks and the high-pressure and low-pressure safety injection pumps. The components and the flow paths are shown in P&ID E-23866-210-130.

The safety injection system injects borated water into the reactor coolant system to increase the shutdown margin during the rapid cooldown following a steam line break. The injection phase is the same as outlined below; however, there is no recirculation phase.

The safety injection tanks and the loop injection valves and manifold are located inside the containment but outside of the bioshield. The safety injection tanks are installed approximately 8 feet above the injection nozzles. The final check valve in each injection line is located close to the reactor coolant pipe. This location provides maximum protection to the reactor coolant system from a rupture in the safety injection piping.

The SIRWT is located at the southwest side of the auxiliary building basement. The safety injection system pumps are located in two watertight rooms at the lowest level of the auxiliary building in a tornado- proof area. This location assures adequate pump suction head when recirculating from the containment. Sufficient space is provided around equipment in these rooms to permit installation of temporary shielding for maintenance. Valves required to isolate equipment are provided with remote operators.

Each safety injection tanks inner check valves can be seated during normal operations using a HPSI Pump. The level of the safety injection tanks can be maintained during normal operations using a HPSI Pump. Boron concentration in the safety injection tanks is adjusted by connecting a temporary chemical addition pump to the SI tank sample line in containment.

The containment has a system of floor drains which conduct water from the containment floors to the containment sump. Consisting mostly of 4 inch pipe, this drain system is shown schematically in P&ID 11405-M-6. In the event of a loss of coolant accident, the floor drains would be severely overloaded. Excess water would flow down an annulus which exists around the periphery of each floor to the basement floor where the safety injection recirculation strainers are located. Adequate openings exist in the steam generator and reactor coolant pump chambers to prevent the accumulation of water (see Figure 1.2-5). Water which falls into the fuel transfer canal will drain through valve WD-883 to the basement floor (see P&ID 11405-M-6). A vertical section showing the ECCS drainage piping from the safety injection recirculation strainer inside the containment to the nearest room in the auxiliary building is shown in Figure 6.2-1. Safety injection pump suction from each line and a cross section of each strainer is shown in Figures 6.2-2 and 6.2-3 respectively.

6.2.4.2 Injection Phase

The borated water in the safety injection tanks is at or above the refueling boron concentration; the tanks are pressurized with nitrogen at a minimum pressure of 240 psig. They are connected to the reactor coolant system cold legs through isolation valves which are normally open. Two check valves prevent reactor coolant from entering the tanks. Injection will occur if the reactor coolant system pressure falls below the combined pressure of the static water head plus the tank gas pressure.

Long term cooling of the core is provided by the safety injection pumps. The safety injection pumps are started automatically by safeguards signals via the sequencer panels (see Section 7.3).

SIAS opens and closes certain valves as shown in P&ID E-23866-210-130. Borated water is initially pumped from the SIRWT to the reactor coolant system. In addition, SIAS aligns the charging pumps in the chemical and volume control system to take suction from the concentrated boric acid storage tanks and starts the boric acid pumps. The USAR Section 14 safety analyses conservatively do not credit this function and this input assumption forms the basis for why the CVCS and charging pumps are not classified as Engineered Safeguards equipment.

6.2.4.3 Recirculation Phase

The recirculation actuation signal (RAS, see Section 7.3.2.7) automatically switches the pump suction to the containment recirculation inlet at elevation 994'-0" when the SIRWT level falls to a preset point. At this time, the flow path from the containment sump is opened, the SIRWT flow path is closed, the low-pressure safety injection pumps are stopped automatically and water is recirculated from the sump by the high-pressure pumps. Water from the containment sump is also circulated by the containment spray pumps and is cooled by the shutdown cooling heat exchangers. The operator may direct a portion of this cooled water to the suction of the high-pressure safety injection pumps. This would provide subcooled water to the core. The low-pressure safety injection pumps may also be used for long term cooling, if the reactor coolant pressure is sufficiently low.

6.2.5 Design Evaluation

The design basis and system requirements during a DBA are met with the operation of three of the four safety injection tanks delivering borated water to the core and with one high pressure injection pump delivering 75 percent of its rated flow to the core and one low-pressure injection pump delivering approximately 75 percent of its rated flow to the core for large pipe breaks. At 30 minutes into a large break LOCA there is sufficient HPSI flow to remove decay heat and keep the core covered with 35% spillage.

The power supplies to safety injection pumps and control valves are arranged so that the loss of one diesel-generator will still permit starting of one low-pressure injection pump and at least one high-pressure injection pump and will also permit opening of one HPSI valve for each of the four coolant loops and two LPSI valves serving two of these coolant loops.

During recirculation, one high-pressure safety injection pump has sufficient capacity to maintain the water level in the reactor vessel above the core. Additionally, a LPSI or CS Pump can be used in accordance with the EOPs, for this function, as containment and reactor coolant system conditions permit.

Ability to meet the core protection criteria is assured by the following design features:

a. A high-capacity passive system which requires no outside source and will supply large quantities of borated water to rapidly recover the core after a major loss-of-coolant accident up to a break of the largest reactor coolant line.

b. A pumping and water storage system with internal redundancy which will inject borated water to provide core protection. The pumping system also provides borated water to keep the core covered and to continue cooling the core after the passive system supply has been injected. In addition, this system will remove reactor core decay and stored heat during long term operation after the reactor coolant system rupture. Instrumentation and sampling provisions allow monitoring of the recirculated coolant.

c. Separated pump rooms and redundant pumping systems which will permit minimum safeguards equipment to operate should one pump room flood in the event of a pipe failure during long term operation.

d. Redundant on-site power supplies in the form of two emergency diesel-generators, each of which has sufficient capacity for minimum safeguards operation.

e. All active components which must function individually for the system's performance to meet the criteria stated for core protection can be tested during normal reactor operation. In addition, extensive shop and preoperational tests were performed to verify adequate component and system operation.

f. Most of the active components are located outside the containment where they are protected from accident-generated missiles and from post-accident environmental conditions. Those active components located inside the containment are shielded by missile barriers and need only operate for a short time period after the accident. Components of the type selected were tested for operation under simulated post-accident conditions.

g. The four injection lines are arranged such that movement of a ruptured reactor coolant pipe will not cause a subsequent failure of injection lines in unruptured loops. The maximum movement of the reactor coolant pipe at the injection nozzle in the unruptured loop will not damage the injection line.

h. The safety injection system has been designed to meet the single failure criterion. This includes the fluid systems and the electrical control and instrument systems.

i. All components, piping, cabling structures, power supplies, etc., in the safety injection support systems are designed to Class 1 seismic criteria as delineated in Appendix F.

The effectiveness of the safety injection system to satisfy the criteria stated for core protection is demonstrated by the blowdown and refill transient curves following a loss-of-coolant accident. This analysis is presented in Section 14.15.

The USAR Chapter 14 loss of coolant accident analysis assumes a volume of 250,000 gallons of water is pumped from the SIRWT when recirculation begins. A minimum useable volume of 283,000 gallons, as required by the Technical Specifications, provides sufficient inventory to account for instrument uncertainty.

The minimum required hydraulic performance for a low pressure safety injection pump is calculated based on the 4-valve LPSI delivery curve used in the LOCA analysis (Ref. 6.5-5). The minimum required hydraulic performance for a high pressure safety injection pump is calculated based on the HPSI delivery curve used in the LOCA analysis (Ref. 6.5-5).

The containment water temperature transients following a LOCA were analyzed with respect to possible adverse effects on the available net positive suction head (NPSH) for the safety injection and containment spray pumps when the system shifted to the recirculation mode. The analyses were calculated based on the same conservative conditions and assumptions utilized in the containment pressure transient analyses as described in Section 14.16 of the USAR. The initial conditions and heat sinks are as listed in Section 14.16.

The containment water temperature transients were calculated assuming that when the coolant flashes, it comes into equilibrium with the containment at the containment atmosphere temperature. The portion of reactor coolant which does not flash drops to the floor as saturated liquid. The initial water mass on the containment floor is reactor coolant from the blowdown (T=0 Sec.) and the contents of one safety injection tank that does not reach the core. No spray or pumped safety injection is assumed until 30 seconds after the accident. Water is added to the sump by spillage from the core, condensation of steam from the containment atmosphere, and by the sprays. The water from all sources is assumed to be at containment atmosphere temperature. With three spray pumps and one spray header available, the containment water temperature from a double ended rupture of a 32 inch diameter reactor coolant pipe is considered for the safety injection mode of USAR Section 14.16. This mode consists of three of the four safety injection tanks, two low pressure safety injection pumps and one high pressure safety injection pump, and no charging pumps.

Assuming the SIRWT is at its most adverse condition, (283,000 gallons of usable water above 16" recirculation level), the following are the system parameters at the start of recirculation:

	Minimum Safety Injection	Full Safety Injection
Recirculation		
Start Time	3740 Sec.	2815 Sec.
Containment Water	172 °F.	174 °F.
Atmosphere	150 °F.	148 °F.

With regard to net positive suction head, the recirculation phase calculation takes credit for suction head provided by containment sump water subcooling due to temperatures less than saturation as calculated in the containment transient analysis. This is an exception from the design criteria set forth in Safety Guide #1; however, the NPSH calculation conservatively credits only 25% of the available sump water subcooling with the conclusion that adequate NPSH is available.

Assuming that only one shutdown cooling heat exchanger is available at recirculation, the spray temperature for minimum safety injection would be 144°F and for full safety injection 145°F.

6.2.6 Availability and Reliability

6.2.6.1 Normal Operation

During normal plant operation, there are no components of the system in operation, with the exception of a HPSI pump that is periodically run to fill an SI tank or for periodic surveillance or maintenance activities. All components are on standby for possible emergency operation.

6.2.6.2 Plant Shutdown

System operation for shutdown cooling is discussed in Section 9.3.

6.2.6.3 Emergency Operation

Safety Injection

The five safety injection pumps (three high-pressure and two low-pressure) are started via the sequencers by PPLS or CPHS. PPLS and/or CPHS also energize the safety injection actuation signal (SIAS), opening the safety injection valves and closing the check valve leakage cooler valves. If all normal power sources are lost and one emergency diesel-generator fails to start, one low-pressure and at least one high-pressure pump are automatically started (see Section 8.4). The rest of the system is aligned for safety injection during power operation. The safety injection tanks will discharge into the reactor coolant system when RCS pressure drops below the minimum SI tank pressure of 240 psig.

Recirculation

When the water in the SIRWT reaches a predetermined low level, the STLS is initiated by coincident low level signals from two of four level switches in the SIRWT. An STLS in coincidence with either a CPHS or PPLS will initiate the recirculation actuation signal (RAS). The RAS opens the containment recirculation valves, closes the SIRWT valves, stops the low-pressure pumps, closes the valves in the pump minimum recirculation lines and cuts in full component cooling water flow to the shutdown heat exchangers. If water is available from the spray pumps and shutdown cooling heat exchangers, a portion of the water discharged from the shutdown cooling heat exchangers may be manually diverted to the high-pressure pump suction. This is a preferred mode of operation, but is not necessary to meet core cooling requirements. The low-pressure pumps may be manually restarted by operation of override switches to obtain increased cooling flow when the reactor coolant system pressure is reduced. One or more spray pumps can also be used to augment flow to the core after the pressure is reduced.

Leaks in the pump rooms during the recirculation mode can be detected by the following:

a. The radiation monitor at the plant ventilation discharge duct;
b. Pump room sump water level;
c. Containment water level;
d. Process flow instrumentation.

Leakage drains to the pump room sump. From there it is pumped to the radioactive waste disposal system (RWDS) for processing. Isolation of the leaking component would be required in the event leakage exceeds the capacity of the RWDS; all equipment in either pump room can be made accessible for maintenance by isolating the affected room and continuing recirculation using equipment in the other pump room. If the discharge duct radiation release approaches permissible limits, the room exhaust air would be passed through a charcoal filter/adsorber (see Section 9.10).

6.2.7 Tests and Inspections

Testing of all components in conformance with the codes and standards referenced in Section 6.2.3 was conducted during fabrication and erection. Qualification testing of the pumps has been performed as described in Section 6.2.3.3. Additional information is given in Sections 1.4.8 and 1.6.4. Routine operational testing of major portions of the logic circuits, pumps and power-actuated valves in the safety injection system is described in Section 7.3.

The pumps are located outside the containment for access and to permit maintenance during normal plant operation. A recirculation line is provided on the discharge of each pump. Periodic testing is performed by recirculating water back to the SIRWT.

In addition to the above tests, the safety injection tank check valves may be tested for leakage during operation. Each safety injection tank has two check valves in series between the tank nozzle and the reactor coolant system. The pressure control system between the check valves is also used to test the check valves. The check valve closest to the tank may be tested by opening the pressure control valve. As the pressure between the check valves decreases, the valve will open under the influence of tank pressure. A flowmeter is provided in the test line to measure flow, and indications of tank level and pressure are available to verify the flow. The discharge of the low pressure safety injection pumps provides the capability of testing the check valve closest to the primary system. Flow from the low pressure safety injection pumps is established. Sufficient flow into the system confirms operation of the check valve. These valves are tested during shutdown with other components of the system to assure their operability.

Safety injection and refueling water tank contents are sampled in accordance with Technical Specifications.

6.3 CONTAINMENT SPRAY SYSTEM

6.3.1 Design Bases

The function of the containment spray system is to limit the containment structure pressure rise thereby reducing the leakage of airborne radioactivity from the containment by providing a means for cooling the containment atmosphere after the occurrence of a loss-of-coolant accident (LOCA).

Pressure reduction is accomplished by spraying cool, borated water into 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 recirculation and cooling system described in Section 6.4 for the containment pressure analysis described in Section 14.16.

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"

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

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).

<u>Shutdown Heat Exchangers, Item No's. AC-4A and 4B</u>

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	350
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 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.

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7.5 INSTRUMENTATION SYSTEMS

7.5.1 Process Instrumentation

7.5.1.1 Design Bases

The non-nuclear process instrumentation measures temperatures, pressures, flows and levels in the reactor coolant system, secondary systems and auxiliary systems. Process variables required for startup, operation and shutdown of the plant are indicated, recorded and controlled from the control room. Other instrumentation which is used less frequently or which requires a minimum of operator action is located near the equipment. Alternate indicators and controls are located at other locations than the main control room to allow reactor shutdown should the main control room have to be evacuated (see Section 7.6).

Four independent measurement channels are provided to monitor each process parameter required for the reactor protective system. Redundant channels are provided for engineered safeguards action to meet the single-failure criterion. The four independent channels provide sufficient redundancy to ensure system action and to allow each channel to be tested during plant operation.

Two channels are provided to monitor parameters required for critical control functions. These channels and associated sensors are independent of the Reactor Protective System, with the exception of the T-121H channel which receives an isolated input signal from the B/T-122H channel.

Redundant subcooled margin monitors are provided to give a continuous indication of margin to saturated conditions. Each of the monitors utilizes two hot leg temperatures, two cold leg temperatures, and a pressurizer pressure signal to calculate the subcooled margin. In addition, the Safety Parameter Display System (SPDS) provides a similar function plus two additional subcooled margins using other temperature signals are calculated.

7.5.1.2 System Description

The process instrumentation described below is associated with the reactor protective, reactor control, reactor plant controls, or reactor instrumentation:

Temperature

Temperature measurements are made with precision resistance temperature detectors (RTD's) which provide a signal to the remote temperature indicating control and safety devices.

The following is a brief description of each of the temperature measurement channels:

a. Hot leg temperature: Each hot leg contains seven temperature measurement channels. Four of these channels provide a narrow range hot leg temperature signal to the thermal margin/low pressure trip circuits and to subcooled margin monitors. Two wide range channels provide signals to the QSPDS for indication and subcooled margin monitoring. The other hot leg temperature measurement channel provides a signal to the loop T_{avg} and W summing computers in the Reactor Regulating system. Channel T-121H receives an isolated T_{hot} signal from channel B/T-122H. The five narrow range hot leg temperatures are indicated on the control panel. A high temperature alarm from each control channel is provided to alert the operator to a high temperature condition.

b. Cold leg temperature: Each cold leg branch contains four temperature measurement channels. Two of the narrow range channels in each branch provide a cold leg temperature signal to the thermal margin/low pressure trip circuits and to the subcooled margin monitors. These channels also provide cold leg temperature indication on the control panel. One wide range channel provides a cold leg temperature signal to the QSPDS for its subcooled margin monitoring function.

The other cold leg temperature measurement channel in one branch of each loop provides a signal to the loop T_{avg} and W summing computers; the remaining cold leg temperature channel provides a signal to the plant computer and is an input to the low temperature/over pressure PORV reduced pressure protection scheme.

c. Loop average temperature: Each loop is provided with an average temperature summer. The T_{avg} summer receives inputs from the control channel hot leg temperature detector and the cold leg temperature detector and provides an average temperature output to the reactor regulating system and to a recorder. Channel T-121H receives an isolated T_{hot} signal from channel B/T-122H. The temperature recorders are equipped with two pens. One pen records the average temperature and the other pen records the programmed reference temperature signal (T_{ref}) corresponding to turbine load (first stage pressure).

d. Loop differential temperature: The loop differential temperature is computed from the control channel hot leg temperature detector signal and the cold leg temperature detector signal. Channel T-121H receives an isolated T_{hot} signal from channel B/T-122H. Each loop differential temperature is recorded in the main control room.

e. There are two subcooled margin monitors, each with a backup channel. The first is a stand alone type using narrow range temperature signals and a pressurizer pressure signal to calculate subcooled margin. The second is the subcooled margin monitor function of the QSPDS. It uses wide range temperature signals and a pressurizer pressure signal to calculate subcooled margin.

Four resistance-thermometer elements, not associated with the reactor protective, reactor control or reactor plant controls, are provided within the containment for temperature indication during DBA and post-DBA conditions. The temperature elements were designed for service at 500°F and 4000 psia.

Additional temperature indication is provided by the Core Exit Thermocouples (CET's) located in the in-core instrumentation (ICI) assemblies.

Level

Reactor vessel water level is provided by the Heated Junction Thermocouple (HJTC) probes. Two probes in the reactor vessel measure the coolant inventory and provide an indication of the collapsed liquid level between the fuel alignment plate and the top of the head. This indication is provided on both the Qualified Safety Parameter Display System (QSPDS) and on the plant Safety Parameter Display System (SPDS).

The HJTC probes contain eight sensors each at varying levels in the reactor vessel. Each sensor senses the existence or absence of coolant at a particular level and the QSPDS microprocessor translates this information to an indication of level of water in the vessel.

Pressure

Pressure is measured by two pressure control channels which are recorded in the control room and provide independent high and low alarms. The transmitter produces a dc current output that is proportional to the pressure sensed by the instrument. The dc current outputs are used to provide signals to the remote pressure indicating, control and safety devices.

The following is a brief description of each of the pressure measurement channels:

a. Pressurizer pressure (protective action): Four pressurizer pressure transmitters provide independent, suppressed range, pressure signals. These four independent pressure channels provide the signals for the reactor protective system high pressure trip and the variable thermal margin/low pressure trip. The channels also provide the low-low pressure signal to initiate safety injection. All four pressure channels are indicated in the control room and high, low, and low-low alarms are annunciated.

b. Pressurizer pressure (protective action): Four pressurizer pressure transmitters in the Diverse Scram System provide independent, high range, pressure signals. These four independent pressure signals are inputs to the Diverse Scram System high pressure trip. Of the four channels, two provide pressure indication in the control room.

c. Pressurizer pressure (control action): Two independent pressure channels provide suppressed range signals for control of the pressurizer heaters and spray valves. The output of either controller may be manually selected to perform the control function.

   The Power Operated Relief Valves are actuated by the high primary system pressure reactor trip signal.

Pressurizer Level

Level is sensed by level transmitters which measure the pressure difference between a reference column of water and the pressurizer water level. This pressure difference is converted to a dc current signal proportional to the level of water in the pressurizer. The dc current outputs of the level transmitters provide signals to the remote level indicating control and safety devices.

Two independent pressurizer level transmitters provide signals for use by the chemical and volume control charging and letdown system. In addition, signals are provided for pressurizer heater override control. These level transmitters are calibrated for steam and water densities existing at normal pressurizer operating conditions.

Each of the two pressurizer level control channels provide a signal for level recorders in the control room. These recorders are two-pen recorders, with one pen recording actual level as sensed by the level control channel and the other pen recording the programmed level set point signal from the reactor regulating system.

Subcooled Margin

The redundant stand alone Subcooled Margin Monitors are an on-line microcomputer-based system which uses reactor coolant process signals to provide a continuous indication of the margin from saturation conditions. These SMM's receive their inputs from hot leg temperature, cold leg temperature and pressurizer pressure loops

The (narrow range) temperature and pressure analog signals are converted to digital signals. These signals are interfaced to a microcomputer. The microcomputer contains steam tables and interpolation routines for which a saturation temperature and pressure are calculated.

By comparing the saturation temperature and pressure to the actual coolant temperature and pressure, a margin from saturation is calculated. Either the temperature or pressure margin can be displayed on the digital panel meter. The margin is also compared to a setpoint for a low margin alarm.

In addition to the above-described Subcooled Margin Monitor with the narrow hot and cold leg temperature input ranges, additional subcooled margin indications are available as part of the Safety Parameter Display System (SPDS). The SPDS performs a subcooled margin calculation similar to the one described above using wide range hot and cold leg temperatures and the same pressure signals as above. A block diagram of the wide range Subcooled Margin Monitor is shown in Figure 7.2-6a. In addition, the SPDS performs two additional subcooled margin calculations. These additional calculations are based on representative Core Exit Thermocouple (CET) temperature, and upper head temperature as sensed by the reactor vessel level Heated Junction Thermocouple (HJTC) probes. All three calculations share the same pressure inputs.

Flow

An indication of reactor coolant flow is obtained from measurement of the pressure drop across each steam generator. The pressure drop is sensed by differential pressure transmitters which convert the pressure difference to dc currents. The dc currents provide a signal to the remote flow indicating and safety devices.

Eight independent differential pressure transmitters are provided in each heat transfer loop to measure the pressure drop across the steam generators. The outputs of one of these from each branch are summed to provide a signal of flow rate through the reactor core, which is indicated and supplied to the reactor protective system for loss-of-flow determination. The differential pressure sensed by each transmitter is indicated in the control room. The arrangement of the flow transmitters is shown in Figure 7.2-5.

Four additional differential pressure transmitters across the reactor coolant pumps and four across the reactor loops furnish additional readings for periodic reevaluation and systems calibration.

7.5.2 Nuclear Instrumentation

7.5.2.1 Design Bases

Ten channels of instrumentation are provided to monitor the neutron flux. The system consists of four wide range logarithmic channels, four power range safety and two power range control channels. Each channel is complete with separate detectors, power supplies, amplifiers and trip units to provide independent operation. The operating capability of the ten monitoring channels is greater than 10 decades of neutron flux and, as such, is more than adequate to monitor the reactor power from shutdown through startup to 200 percent of full power. Channel range, sensitivity and overlap are shown in Figure 7.2-8.

The neutron flux detectors are located in instrument thimbles in the biological shield around the reactor vessel. The power range control channel detectors are placed 180 degrees apart. The power range safety channel detectors are placed in thimbles approximately equally spaced around the reactor vessel as shown in Figure 7.5-1.

7.5.2.2 System Description

The nuclear instrumentation system consists of ten independent channels.

Installation

The nuclear instrumentation signal processor is located in the reactor protective system cabinet in the control room (see Section 7.6.2). Four cabinets designated as A, B, C and D each house one channel of the protective system. Each cabinet contains one power range safety channel and one wide range logarithmic channel. Mechanical and thermal barriers between the cabinets reduce the possibility of common event failure. The detector cables are routed separately from each other. This includes separation at the containment penetration areas. The nuclear detector locations are shown in Figure 7.5-1.

Functional Description

Ten channels of instrumentation are provided to monitor the neutron flux. The system consists of wide range logarithmic channels, power range safety and power range control channels. Each channel is complete with separate detectors, power supplies, amplifiers, and bistables to provide independent operation. The operating capability of the ten monitoring channels is greater than 10 decades of neutron flux and is more than adequate to monitor the reactor power from shutdown through startup to 200 percent of full power.

Four wide range logarithmic channels monitor the flux from source level to above full power. The flux signals, obtained from a dual fission chambers, are amplified and transmitted to the signal processors located in the control room. Audible count rate signals are available in the control room. In addition to the information on the neutron flux, these channels provide a rate-of-change-of-power signal to the reactor protective system for CEA withdrawal prohibit or reactor trip.

Four channels are designated as power range safety channels and provide signal outputs to the reactor protective system. These channels operate from 0 to 200 percent of full power. Power level signals from these channels are supplied to the protective system. These four channels contain detectors composed of dual section ion chambers which monitor the full axial length of the reactor core at four circumferential positions. They can also detect axial flux tilt.

Two separate power range control channels, which are identical to the power range safety channels, provide reactor power signals to the reactor regulating system. The channel output is a signal directly proportional to reactor power from 0 to 200 percent.

The gain of each channel is adjustable to provide a means for calibrating the output against a plant heat balance. Each channel provides a power reference signal to one of the independent reactor regulating system channels.

7.5.2.3 Design Criteria

The system was generally designed in accordance with the criteria of IEEE 279, August 1968. In areas not covered or specifically identified by the criteria, the following criteria were used:

a. The nuclear instrumentation sensors are located so as to detect representative core flux conditions;

b. Multiple channels are used in each flux range;

c. The channel ranges overlap sufficiently to ensure that the flux is continually monitored from source range to 200 percent of rated power;

d. The power range safety channels are separate from, and independent of, the control channels;

e. Each of the four power range safety channels is physically segregated from the others. Each of the four startup and logarithmic channels is physically segregated from the others;

f. Power is supplied to the system from four separate ac buses. Loss of one bus trips one safety channel and one startup or logarithmic channel;

g. Loss of power to channel logic results in a channel trip;

h. All channel outputs are buffered so that accidental connection to 120-Volts ac, or to channel supply voltage, or shorting individual outputs has no effect on any of the other outputs.

7.5.2.4 Wide Range Logarithmic Channel Description

The four (4) wide range nuclear instrumentation channels provide relative indication of the neutron flux level at the detector assemblies and the measure of the rate of change of neutron flux from source level to above 100% of full power. Dual fission chambers are used to detect the neutron flux. The signal from the fission chambers is provided to an amplifier assembly where the signal is amplified with some conditioning. The monitor in the main control room provides further processing of the detector signal into parameter signals.

The monitors provide indication of source range level (CPS), wide range level (% power) and rate of change of power (DPM). These signals are provided as outputs for remote indication. LED lamps provide status indication of the two adjustable bistables, A/C power, Channel in Test, and a channel fault condition. The rate of change of power signal is utilized as an input to a trip unit within the Reactor Protective System (RPS). The RPS trip unit pretrip signal provides an input to the CEA Withdrawal Prohibit function. The source range level signal is provided as an input to the Audio Countrate drawer.

The two adjustable bistables monitor the wide range power level signal. One bistable serves to enable Zero Power Mode Bypass for the Reactor Protective System (RPS) and to disable the rate of change of power signal provided to the RPS. The second adjustable bistable provides a contact input to the SCEAPIS for lower power cutout and annunciation input.

Any one of the four (4) channels of wide range nuclear instrumentation can be selected on the Audio Countrate drawer as the input to the Audio Countrate circuitry. The Audio Countrate drawer contains a local speaker and provides an output to a remote speaker in containment.

One channel of wide range nuclear instrumentation provides input signals to a separate monitor at the alternate shutdown panel location (Section 7.6.4). This monitor and associated amplifier assembly are isolated from the main control room signal processor to provide independence in the event of a main control room fire.

7.5.2.5 Power Range Safety Channel Description

The four power range channels measure flux linearly over the range of 1 percent to 200 percent of full power. The detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The upper and lower sections have a total active length of 12 feet. The dc current signal from each of the ion chambers is fed directly to the control room drawer assembly without preamplification. Integral shielded cable is used in the region of high neutron and gamma flux.

The signal from each ion chamber (sub-channel) is fed to an independent amplifier. Within each channel the outputs of the two amplifiers are indicated and averaged. The averaged output of the two amplifiers forms the channel output and feeds remote recorders, bistables and the reactor protective system. In addition, each of the four channel outputs is averaged in a comparator averager module to form an overall average of the four power range safety channels. This overall average is compared to each ion chamber sub-channel output to provide a deviation signal. When this deviation reaches a manually adjustable setpoint, an alarm will annunciate on the main control board.

The 1.0 to 200 percent full scale output is always fed to a recorder and to the bistable which is used by the reactor protective system to disable the wide range logarithmic channel rate trip above 15 percent full power. The summing circuit also has an X2 gain selector switch which disconnects the input of one ion chamber and doubles the gain for the other ion chamber in order to allow full scale power indication with one inoperative ion chamber.

The 1.0 to 200 percent full scale signal is also fed to a CEA drop detection circuit which compares the present value of the signal with the time delayed (5 to 15 seconds) signal level.

Channel calibration and test is accomplished by an internal current source which checks amplifier gain and linearity. A check of the high flux trip setpoint is provided by a current signal which is added to the normal detector output.

Each power range channel contains two bistables. One, previously mentioned, disables the rate of change of power trip signal, as well as enables the axial power distribution trip and the loss of load trip signals; the other, a failure monitor, initiates an alarm on decrease of detector voltage, drawer calibration, or removal of any of the drawer modules. The condition of each bistable is shown by a front panel light.

7.5.2.6 Power Range Control Channel Description

The power range control channels are identical to the power range safety channels. They are located on the reactor protective system nuclear instrumentation cabinet AI-31E in the main control room. These power level signals are connected only with the reactor regulating system and to remote meters.

7.5.3 CEA Position Instrumentation

7.5.3.1 Design Bases

There are 41 control element drive mechanisms (CEDM's). Four are spares. The remaining 37 control element drive mechanisms (CEDM's) are each equipped with two indication systems to provide the operator with position information. The primary and secondary CEA position sensing systems are separate and independent.

The primary CEA position indication system utilizes the output of a synchro transmitter geared to the clutch output shaft. CEA position is displayed visually at the main control panel. One position indicating meter is provided for each group; any CEA within the group may be selected for monitoring. The position of all CEA's may be printed by the MODCOMP computer printer on demand at any time. CEA position information is also used to initiate alarms when limiting conditions are approached, to provide contact closures for sequencing and control, and to monitor for an alarm position deviation between individual CEA's within a group. During a drop test, the system measures and records the time for a CEA to reach the 90 percent inserted position after the clutch is released.

The secondary CEA position system utilizes the output of a voltage divider network controlled by a series of reed switches. The reed switches are actuated by a permanent magnet attached to the rack assembly. Position information is supplied to a cathode ray tube position indicator for simultaneous viewing of all CEA group positions. Individual control rod groups are displayed when selected by the operator. During a rod drop test, the system measures and records the time for a CEA to reach the 90 percent inserted position after the clutch is released.

7.5.3.2 Primary Position Indication System Description

The primary CEA position sensing system determines CEA position by use of synchros. Outputs are provided for visual display and for CEA control. The systems major components are:

a. Thirty-seven CEA position synchro transmitters;

b. Seven visual displays, one for each group, and each with one switch to select any CEA within a group. The displays are synchro receiver indicators and are driven directly by the selected synchro transmitter;

c. The MODCOMP computer and its output printer. The computer receives information directly from the synchro transmitters and is fully independent of the visual displays.

The synchro transmitter for each CEA is geared to the CEA drive shaft below the CEA clutch. Full CEA travel corresponds to 264 degrees of synchro rotation. Synchro output is transmitted to the display receivers and independently to the plant computer which scans and converts synchro outputs into inches of CEA withdrawal. The resolution of this system is approximately $\pm$0.5 inch.

The operator has two means of determining CEA position from the primary CEA position sensing system:

a. Seven visual displays (dials) are mounted above the CEDM controls on the main control panel. There is one display for each group; a selector switch at each display allows the position of any CEA in that group to be indicated. Should any CEA's within the group deviate in position more than a preset amount from any other in the group, a deviation alarm alerts the operator. The out-of-limits CEA can be identified by observing the printout or by checking the position of each CEA within a group by the visual displays.

b. A printer connected to the MODCOMP computer can be used to print out any or all positions either hourly or upon request by the operator.

7.5.3.3 Secondary Position Indication System

The secondary position sensing system measures CEA position by use of magnetic reed switches actuated by a permanent magnet attached to the rack assembly. Simultaneous visual display of all CEA groups is provided by a cathode ray tube position indicator. Indication of individual control rod position is provided on the cathode ray tube display when a particular control rod group display is selected by the operator. The secondary position sensing system is mechanically and electrically isolated from the primary position sensing system. The resolution of the secondary system is approximately ±2 inches.

The system contains:

a. Thirty-seven reed switch/resistor assemblies;

b. The cathode ray tube display unit and associated electronics.

An assembly containing a number of series resistors to form a voltage divider network with reed switches connected at each junction is attached to the CEA extension housing. A voltage is applied to the network; output voltage depends on which reed switches are closed in the voltage divider. A magnet on top of the CEA extension actuates the reed switches as the CEA moves. Overlap between adjacent reed switches is provided. The output is a voltage directly proportional to CEA position.

The outputs from all assemblies are sent to the cathode ray tube display unit. This unit is completely independent of the primary position system.

The Secondary Position Indication System monitors control rod position and reactor power and initiates alarms when the following abnormal CEA configurations are detected:

a. CEA Deviation

b. CEA Regulating Group Overlap

c. CEA Regulating Group Out-of-sequence withdrawal/insertion

d. CEA Insertion to the Pre-Power Dependent Insertion Limit

e. CEA Insertion to the Power Dependent Insertion Limit

f. Regulating Group Withdrawal Prohibit (ISH)

g. Shutdown Group Insertion Permissive (IRG)

The Secondary Position Indication System also performs a rod drop timer function and initiates the actuation of the Rod Block System.

7.5.3.4 Rod Block System

The rod block system is automatically initiated by the Secondary Position Indication System to inhibit all CEA motion in the event a Limiting Condition for Operation (LCO) on CEA insertion, CEA deviation, CEA overlap or CEA sequencing is approached.

The installation of the rod block system ensures that no single failure in the control element drive control system (other than a dropped CEA) can cause the CEA's to move such that the CEA insertion, deviation, sequencing or overlap limits are exceeded. Accordingly, with the rod block system installed, only the dropped CEA event is considered an AOO and factored into the derivation of the Limiting Safety System Settings and Limiting Conditions for Operation.

7.5.4 In-Core Instrumentation

7.5.4.1 Design Bases

The primary function of the in-core instrumentation is to provide measured data which may be used in evaluating the neutron flux distribution in the reactor core. This data may be used to evaluate thermal margins and to estimate local fuel burnup.

The bases for the design of the in-core monitoring system are as follows:

a. Detector assemblies are installed in the reactor core at selected locations to obtain core neutron flux and coolant temperature information during reactor operation in the power range;

b. Flux detectors of the self-powered type, with proven capabilities for in-core service, are used;

c. The information obtained from the detector assemblies is used for fuel management purposes and to assess the core performance. It will not be relied on for automatic protective functions;

d. The output signal of the flux detectors is calibrated or adjusted for changes in sensitivity due to emitter material burnup;

e. Each detector assembly is comprised of four local neutron flux detectors stacked vertically for axial monitoring, and one thermocouple at the assembly outlet.

Axial spacing of the detectors in each assembly and radial spacing of the assemblies permit representative neutron flux mapping of the core and monitoring of the fuel assembly coolant outlet temperatures.

7.5.4.2 System Description

The in-core instrumentation system consists of 28 fixed in-core detector assemblies inserted into selected fuel assemblies. Each assembly contains four rhodium detectors, and one thermocouple. Outputs may be read on the terminals and printers in the control room. These units with their cabling are contained inside an Inconel sheath.

Assemblies are inserted into the core through six instrumentation ports in the reactor vessel head. Each assembly is guided into position in an empty CEA tube in the center of the fuel assembly via a fixed stainless steel guide tube. The seal plug forms a pressure boundary for each assembly at the reactor vessel head as does the Gralock adaptor hub to the reactor vessel flange assembly.

The neutron detectors produce a current proportional to neutron flux by a neutron-beta reaction in the detector wire. The emitter, which is the central conductor in the coaxial detector, is made of Rhodium 103 and has a high thermal neutron capture cross section. The rhodium detectors are provided to measure flux at four axial locations in the fuel assemblies.

The data from the detectors are read by the Emergency Response Facilities (ERF) plant computer which scans all assemblies and prints out the data periodically or on demand. The computer continually computes integrated flux at each detector to update detector sensitivity factors to compensate for detector burnout.

7.5.4.3 ICI Requirements for Monitoring Technical Specifications

On July 16, 1993, the USNRC issued a Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors. The Final Policy Statement contains four criteria which can be used to determine which constraints on the design and operation of nuclear power plants are appropriate for inclusion in the plant's Technical Specifications. The ICI System does not meet any of those four criteria. Subsequently, on February 10, 1995, OPPD requested the elimination of Technical Specification 2.10.3 and the relocation of the Technical Specification limitations on the use of the ICI System to the Fort Calhoun Station Updated Safety Analysis Report (USAR). The USNRC issued a Safety Evaluation Report (SER), dated June 26, 1995, approving OPPD's request (Reference Amendment No. 167). The SER stated that in order to change the requirements concerning the number and location of functional detectors, a successful 10 CFR 50.59 safety evaluation with a rigorous evaluation and justification is required. The following considerations must be included in a 10 CFR 50.59 evaluation if changes to the ICI System requirements are proposed:

1) How an inadvertent loading of a fuel assembly into an improper location will be detected,

2) How the validity of the tilt estimates will be ensured,

3) How adequate core coverage will be maintained,

4) A list of the measurement uncertainties and why the added uncertainties are adequate to guarantee that measured peak linear heat rates, peak pin powers radial peaking factors, and azimuthal power tilts will meet TS limits, and

5) How the ICI System will be restored to at least 75 percent prior to the beginning of a new cycle.

The following information represents the ICI requirements for measuring Technical Specification values:

The ICI System shall be operable with either:

1) At least 75% of all incore instruments and a minimum of two incore detector strings per full axial length quadrant whenever the ICI System is used to monitor the planar radial peaking factor (F_{xy}^T), the intergrated radial peaking factor (F_R^T), the total peaking factor (F_Q^T), the radial power distribution, the peak linear heat rate, and the azimuthal power tilt, or

2) At least 28% but less than 75% of all Incore Detector Strings and:

- At least two Incore Detector Strings are operable per Axial Quadrant whenever the ICI System is used to monitor the planar radial peaking factor (F_{xy}^T), the integrated radial peaking factor (F_R^T), the total peaking factor (F_Q^T), the radial power distribution, the peak linear heat rate, and the azimuthal power tilt, and

- An increase of 1% to the total uncertainties applied to the planar radial peaking factor (F_{xy}^T), the integrated radial peaking factor (F_R^T), and the total peaking factor (F_Q^T), (Reference 7.7.16), and

- The frequency of determining total integrated and planar radial peaking factors is changed to a minimum of once every 15 days.

An operable in-core instrument shall consist of three or more operable rhodium detectors.

A quadrant symmetric in-core instrument location consists of a location with a symmetric counterpart in any other quadrant.

Following each fuel loading:

- The ICI System must have at least 75% of the in-core instruments operable, and

- The initial measurement of the linear heat rate, F_{xy}^{T}, F_{r}^{T} and azimuthal power tilt shall consist of the first full core power distribution calculation based on in-core detector signals made at a power level greater than 40 percent of rated power.

For recalibration of the ex-core detectors, a minimum of four in-core instrument locations at each detector level (or a total of 16 detectors) with at least one location in the center seven rows of fuel assemblies and at least one location outside the center seven rows of fuel assemblies shall be operable.

With the ICI System inoperable, do not use the ICI System for 1) recalibration of the ex-core detector inputs to the axial power distribution trip calculator, and 2) monitoring of peak linear heat rate and radial power distribution.

The linear heat rate shall not exceed the limits of the Allowable Peak Linear Heat Rate vs. Burnup Figure provided in the COLR when the following uncertainties are appropriately applied:

- A flux peaking augmentation factor as shown in Technical Specification Figure 2-8,

- A measurement calculational uncertainty factor of 1.062 for more than 75% of the ICIs operable and 1.072 for between 75% and 28% of the ICIs operable,

- An engineering uncertainty factor of 1.03,

- A linear heat rate uncertainty factor of 1.002 due to axial fuel densification and thermal expansion, and

- A power measurement uncertainty factor of 1.013 at 100% power, 2.1% at 65% power and 5% below 30% power (Ref. 7.7.17).

A statistical combination of the above uncertainties (SCU) yields a 1.09 multiplier for more than 75% of the ICIs operable and a 1.11 multiplier for between 75% and 28% of the ICIs operable to the peak linear heat rate measurement. This SCU analysis (Ref. 7.7.18) considered the effects of a variable power measurement uncertainty and fuel rod bowing effects on peak linear heat rate.

7.5.5 Plant Computer (ERF System)

7.5.5.1 Design Bases

The Emergency Response Facilities (ERF) System is used to monitor and log plant parameters and equipment status, to perform some secondary plant performance calculations and to provide the primary Safety Parameter Display System (SPDS).

The principal purpose and function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. A Human Factors Maintenance Plan has been and will continue to be used for SPDS display formats and techniques.

The computer is not part of the Reactor Protective System. The function of the computer is to assist the operator in optimizing plant performance and assimilate plant data. The computer provides both a record of the operation of the plant and a means of readily providing the operator with information of the following type:

a. The current status of certain plant switches, relay contacts, values, or plant parameters;

b. The displayed value or trend of selected plant parameters or calculated values;

c. Post-trip review of selected plant analog parameters.

7.5.5.2 System Description

The computer is a real time digital processing system which collects and organizes plant data for reference and display in the Control Room, Technical Support Center (TSC) and in the Emergency Operations Facility (EOF). The ERF computer is a distributed MODCOMP based system which consists of dual processors for data acquisition and dual processors which serve as the host for the system.

User interaction with the computer is through the host. The arming plans are established through user interface to the host. Display selections and other system options are made through the host user interface. The host controls storage of on-line history files, audit trails, and transient data files. It performs other system functions, such as initial program loading for the Data Acquisition System (DAS), security protection, and communication with external data systems.

The system provides supplementary information for assistance in plant operation and provides logic for automatic identification and alarm of off-normal conditions.

The computer interfaces with the Inadequate Core Cooling Instrumentation (ICCI) System via fiber optic data links. The ICCI System provides numerous channels of measurement data and valve position indication to the computer data base for processing and display.

The system scans numerous analog inputs (flow, temperature, pressure and level), and digital inputs (including the position of valves, relay contacts, circuit breakers and switches). It provides indication and alarm of off-normal conditions on the system terminals and printers.

An analog input can be simultaneously selected for direct observation on a two pen recorder; two such recorders are provided.

The plant design includes sufficient instrumentation to permit safe operation of the plant at all times, irrespective of computer availability.

The SPDS displays and software also reside in the system. The variables displayed on the SPDS screens were chosen to aid operators in the diagnosis and mitigation of transients and accidents. The parameters necessary for evaluation of the events of the Emergency Operating Procedures and their Safety Function Status Checks are located on these 7 mid-level screens or the associated screens available from these displays. The parameters included for display are consistent with those contained in References (1) and (2) and are grouped under the EOP Safety Functions as listed below:

a. Reactivity Control

b. Vital Auxiliaries

c. RCS Inventory Control

d. RCS Pressure Control

e. Core Heat Removal

f. RCS Heat Removal

g. Containment Integrity

7.5.5.3 Terminal/User Interface Description

The system contains numerous terminals to provide communication between the user and the computer. The computer executes various functions in response to commands entered at the terminals.

7.5.5.4 Program Functions

<u>Scanning and Display</u>

The computer system checks digital inputs from valves, relay contacts and position switches and initiates an alarm in the event of an improper change in status.

The system also checks analog inputs for the following:

a. Operability of sensors and instrument transmitters. Signal in proper range.

b. Comparison of sensor output to alarm setpoints for alarm initiation.

The SPDS software takes information supplied by plant sensors and uses it to develop mid- and top-level graphic displays that provide an indication of the plant safety functions. Selected combinations of plant sensors and computed parameters are used to drive the mid-level bar graph displays. This information is then further processed into a top-level display which serves as an "annunciator" for the safety functions and their alarms.

<u>Calculations</u>

The system performs functions necessary for conversion of inputs to engineering units and preparation of some plant performance calculations (linearization of thermocouple inputs, square root extraction, integration and averaging).

The system also detects flux tilting factors by using the out-of-core nuclear instruments. This program calculates a single magnitude and angle for the excore Quadrant Power Tilt Detection System. The program has logging capabilities which include an hourly and demand log of tilt (%), angle in degrees and harmonic indexes for safety channels A through D and control channels A and B. Vector averaging is also included on these logs for the detectors.

R0002X	Control A Linear
R0002Y	Control A Linear
R0002X	Control B Linear
R0002Y	Control B Linear

The computer performs functions necessary for conversion of inputs to engineering units and preparation of the plant performance calculations, e.g., linearization of thermocouple inputs, scale factoring, zero-suppression, square root extraction, integration and averaging.

A heat balance is performed by computing the enthalpy rise in the steam generator secondary side, correcting for blowdown and stored heat in the steam generator. The reactor thermal output is computed by summing the thermal outputs from each of the steam generators (secondary side), correcting with programmed constants to compensate for the heat input from reactor coolant pump and pressurizer operation and for losses, e.g., heat losses to ambient. Finally, an on-line comparison is made between the calculated reactor output and all four power range nuclear instrument channels. Alarms are initiated if the relative readings differ by more than a predetermined percentage of indicated full power.

The calculated thermal output of the two steam generators is summed, and the fraction of the total output contributed by each steam generator is computed to obtain a thermal tilting factor.

Performance Logging

The computer system logs the position of all CEA'S and alarms the deviation in position of an individual CEA within a group. The integrated power history for each CEA in specified core axial intervals is also logged.

A plant performance log is automatically printed out each hour. The data logged normally includes the plant operating parameters such as temperature, pressure and reactor coolant flow, and computed values of reactor thermal output and steam generator output. A selected log of these items is also printed out daily.

CEA Position

The plant computer functions include:

a. Monitoring the CEA synchros and checking the CEA position against limiting positions;

b. Initiating alarms and interlocks under certain limiting CEA positions (CEA's at upper and lower stops, regulating and shutdown CEA insertion limit alarm);

c. Providing contact outputs under other CEA positions (these outputs are used as permissive conditions in the sequencing controls);

d. Checking positions of all CEA's within a given group for deviations in position, and alarming if the deviations exceed a preset value;

e. Printing out all CEA positions hourly;

f. Calculating the CEA drop time.

In-Core Instrumentation

The in-core flux detectors are scanned, corrected for background and sensitivity factors, and converted into thermal neutron flux and power readings. These data are available for display when it has been requested and are logged automatically. The system also collects in-core data and operating parameters (power levels) required for offsite fuel management studies.

Sequence-of-Events Monitoring

The Sequence-of-Events (SOE) hardware and software are used to provide event sequencing of selected digital inputs. The SOE log which prints automatically to a preselected device provides one of the tools for determining that the plant can be restarted safely.

The SOE System monitors inputs from the Reactor Protective System (RPS) trip units and from the diesel safety panels (AI-30A and AI-30B) which perform the automatic load sequencing for the Engineered Safety Features (ESF) equipment.

Emergency Response Data System

The Emergency Response Data System (ERDS) hardware and software is used to provide a data link with the NRC. The data link provides pre-selected plant parameter information to the NRC when an emergency classification is declared in accordance with 10 CFR 50, Appendix E, Section VI. The ERDS is manually initiated. Once the link is established no additional manual actions are required.

7.5.5.5 MINI-CECOR/BASSS

MINI-CECOR is a NRC approved computer program installed on the Plant Computer System, which synthesizes three-dimensional assembly and peak pin power distributions from fixed in-core detector signals. These distributions are useful for monitoring reactor operation with respect to Technical Specification and COLR limits.

In-core detector signals and general plant operating data are input to MINI-CECOR via the Plant Computer System. Detailed three-dimensional assembly burnup distributions are input from an exposure computer file. General reactor data, configuration, fuel characteristics and geometry data are input from a geometry file. Using all of this data, the signals are synthesized into three-dimensional assembly and peak pin power distributions. Power to signal ratios convert the in-core signals (from instrumented assemblies) to assembly powers, while coupling coefficients translate signals from instrumented assemblies to uninstrumented assembly powers.

BASSS (Better Axial Shape Selection System) calculates thermal margin using the output of MINI-CECOR and actual plant conditions to gain operating margin. Specifically, BASSS uses power level, rod insertion, and unrodded radial peaking factors to produce DNB limits as a function of axial shape index (ASI) and power. BASSS then selects the most appropriate ASI and power combinations for the current conditions and produces the maximum allowable amount of operating margin for those conditions.

7.5.6 Inadequate Core Cooling Instrumentation

7.5.6.1 Design Bases

The Inadequate Core Cooling Instrumentation (ICCI) system is used to monitor selected plant parameters and equipment status to detect the approach to, existence of, and recovery from an inadequate core cooling situation. The safety grade processing and display of this information is performed by the Qualified Safety Parameter Display System (QSPDS). The QSPDS also provides safety grade processing and display of selected Regulatory Guide 1.97 variables and accident monitoring instrumentation. The QSPDS serves as a safety grade to non-safety grade isolator for transmission of these class 1E signals to the ERF computer system.

7.5.6.2 System Description

The ICCI system is comprised of the following major pieces of equipment (see Figure 7.5-2):

a. Heated Junction Thermocouple Probe (HJTC). The HJTC's (two redundant sensors) provide an indication of liquid inventory in the reactor vessel above the core. It uses heated and unheated thermocouple junctions at discrete elevations in the reactor vessel to detect the presence of liquid. When the liquid level drops below the heated junction, the less favorable heat transfer characteristics result in an increased temperature of the heated junction without a corresponding change in the temperature of the unheated junction. The temperature difference between the heated and unheated junctions is used as an indication of liquid level by indicating whether or not the thermocouple is uncovered.

b. The 28 Core Exit Thermocouples (CET's) are situated at selected locations in the reactor vessel within each in-core instrument (ICI). The CET's are located above the core and provide an indication of the temperature of the coolant as it leaves the active core region.

c. The HJTC and CET signals are transmitted in containment via mineral insulated (MI) cables. The MI cable is a silicon dioxide insulated multi-conductor stainless steel sheath cabling which is hermetically sealed through an all-welded construction technique. A grounded copper inner sheath is used to block electromagnetic interference. The cables are terminated in multi-pin connectors. The MI cables carry the signals from the instruments to the electrical penetrations.

d. The Qualified Safety Parameter Display System (QSPDS) provides the signal processing and display for the ICCI variables. The QSPDS utilizes a micro-processor based design for the signal processing equipment in conjunction with a display having alphanumeric representation and associated keyboard for each of the two channels. Each channel accepts and processes input parameter signals and transmits its output to the plasma display unit (PDU), an alphanumeric display device. In addition, each channel transmits its output to the ERF computer system.

7.5.6.3 Program Functions

The QSPDS displays, in converted engineering units, all ICCI inputs. These variables are arranged on various display pages, grouped by common critical functions. A hierarchy of displays is utilized to facilitate ease of use and to direct the operator to the pertinent pages which may be in an alarm status.

In addition to the display of the ICCI input variables, the QSPDS also performs the following functions:

a. Bad Data Checking

b. Range Checking

c. Alarm Setpoints and Deadbands for Variables

d. Calculation of Derived Variables:

- RCS Saturation margin
- Representative CET Temperature
- CET Saturation Margin
- Reactor Vessel Level
- Upper Reactor Head Temperature
- Highest/Next Highest CET per Quadrant

e. Flagging of Suspect Calculations

f. Control of Power to HJTC Heaters

g. Transmission of Data to the ERF Based Computer (including the SPDS)

In addition to the ICCI variables, the QSPDS accepts, processes, displays and transmits to the ERF computer other Reg. Guide 1.97 variables. These include both analog (e.g., steam generator levels and pressures) and digital (i.e., critical containment isolation valve positions) indications.

7.5.6.4 Testing

The ICCI and QSPDS computer displays are verified by checking the values against hand calculations using standard steam tables. The HJTC vessel inventory is verified by hand calculations. The Core Exit Thermocouple readings are checked for validity.

7.6 OPERATING CONTROL STATIONS

7.6.1 General Layout

Most of the operating control stations described in Section 7.6 are, in whole or in part, associated with Engineered Safeguards Control and Instrumentation Systems. Some panels are solely devoted to Engineered Safeguards equipment, while others contain both Engineered Safeguards and non-Engineered Safeguards equipment. See Section 6.1.2.3 for definition of Engineered Safeguards control and instrumentation systems. For more supporting detail of the Engineered Safeguards components housed in the various control panels, the design documents, Technical Specifications, CQE Manual, EEQ Manual and Regulatory Guide 1.97 Responses should be consulted.

Normal plant control, including startup, shutdown and normal operation, is handled in the control room. This room is located in the northeast corner of the auxiliary building at elevation 1,036'-0". The room is adjacent to the operating floor of the turbine building.

Other control functions are handled at the following locations:

a. The waste disposal system is controlled from panels AI-100, AI-102, and AI-103, located in the auxiliary building on the north side at elevation 989'-0".

b. A local control station, panel AI-179, for the steam driven auxiliary feedwater pump and associated valves is located in the auxiliary building in the upper electrical penetration room. This panel is used to control the steam generator level if the control room cannot be occupied and auxiliary feedwater is required.

c. Local control stations for the diesel-generators, panels AI-133A and AI-133B, are installed in the diesel-generator rooms. The diesel-generators can be started locally and controlled from these panels in the event that the control room cannot be occupied. Diesel-generator loads can be controlled from switchgear and control centers under such circumstances.

d. The auxiliary building area ventilation air temperatures are controlled at local panels adjacent to the air handling units.

e. Air compressors are controlled from a local panel. This panel includes sequencing controls.

f. The following local panels provide for operation of steam plant equipment:

1. Condenser backwash (AI-123);
2. HP heater drains (AI-121);
3. Screen wash (AI-120);
4. LP heater drains (AI-122);
5. Stator winding cooling and hydrogen control cabinet (AI-134);
6. Alternate shutdown panel (AI-185).

The turbine can be tripped at the turbine standard.

g. The demineralized water plant is controlled from panel AI-104, located in the turbine building plant area.

h. The vacuum deaerator in the demineralized water system is located in the auxiliary building and is controlled from panel AI-105 which is near the deaerator at elevation 1,025'-0".

i. Control of reactor coolant system chemistry (i.e., demineralizers in service and pH chemical addition) is local.

j. The sampling system panels are local to the sampling equipment; these panels are described in Section 9.13.

7.6.2 Main Control Room

The location of the main control room is shown in P&ID 11405-A-8. This location places the control room in an area designed as Class I with regard to seismic disturbances and tornadoes. It also provides easy physical access to the turbine room and the reactor plant. The plant has been laid out to minimize the length of cable run to the switchgear. A cable spreading room is located directly underneath the control room. The arrangement inside the control room is shown in Figure 7.6-1.

The main control board is a duplex benchboard. The equipment requiring the most immediate attention from the operator is located near the center, with those items less likely to require his attention located toward the edges. Panel CB-4, located in the corner of the L, is the reactivity control panel. The primary and secondary CEA position indicators are here, along with the CEDM controls. Controls for boric acid concentration are also located on this panel.

Directly to the right of panel CB-4 is panel CB-10/11, which contains steam generator, turbine, and other steam plant controls. Directly adjacent to panel CB-4, are the controls used to regulate the steam generators, to establish equilibrium conditions in the reactor coolant system and steam generators while going critical, and to operate the auxiliary feedwater and steam dump and bypass systems. This area includes steam generator level indicators, auxiliary and main feed pump and condensate controls and supervisory instruments, and steam dump valve controls. This grouping allows the operator to concentrate his attention on a relatively small portion of the control board during plant startup and following a plant trip. Further to the right, on the same board, are the turbine operation and supervisory instrumentation. The plant electrical output is regulated from this part of the control panel.

Reactor plant and auxiliary controls are on panel CB-1/2/3, located to the left of panel CB-4. Immediately next to panel CB-4 are the reactor coolant system controls and supervisory instruments, including reactor coolant loop flow and temperature indicators. Next are the pressurizer controls and supervisory equipment, including pressure and level indicators and controls, and relief and spray valve switches and controls. Next in order are the chemical and volume control system controls. Normally, this system operates automatically. The operator can override automatic operation if necessary. Next in sequence are the controls for the shutdown cooling system. This system is used only during shutdown and refueling and is manually controlled. The left-hand third of this panel is occupied by the component cooling and raw water system controls. These systems also operate automatically, and the operator can override the automatic controls. The controls located here govern pump operation and the valves which determine the flow paths used by the coolant. The latter capability is given the operator so that he can maintain control of the system in the event of malfunctions. System trimming, to meet flow requirements, is controlled from panel AI-45, behind the main control board.

The extreme right-hand end of the main control board is formed by the electrical panel CB-20. The electrical distribution system within the plant and the electrical connections between the plant and external electrical systems are controlled here. Synchronization of the generator is controlled from the area of the board adjacent to the turbine board. Once set up, this board requires minimal operator attention during normal plant operation.

Computer "desks" are arranged in the area directly before the main control boards. These desks include the operator's console with equipment and user terminals for communication with the computers and provision for writing areas.

A row of panels across the room from the main control board, contains the nuclear instrumentation system and safeguards controls. These are described in order from left to right looking at the panels.

On the left are panels AI-43A and AI-43B, the containment isolation panels. Each panel is in two sections. They incorporate position indicating lights for all containment isolation valves and control switches for all isolation valves which do not have control switches elsewhere in the control room. If a valve fails to close upon the containment isolation signal the operator can remote-manually close the valve from their panels. If a given line has two isolation valves, one is on each panel. A description of the containment isolation system is contained in Section 5.9.5. Containment isolation is a part of Engineered Safeguards as defined in Section 6.

Panels AI-30A and AI-30B are the Engineered Safeguards control panels; these panels are described in Section 7.3.3.

Next in line is the reactor protective system nuclear instrumentation panel, AI-31A through E. Each channel of nuclear instrumentation is equipped with a power level indicator.

The last cabinets in this line are the radiation monitoring panels AI-33A, B, and C. These panels have alarms, indicators, and recorders for the radiation monitors.

Panels AI-65A & B and AI-66A & B, located in the N. corner of the Control Room are Post Accident Monitoring Panels and are instrumented to annunciate, monitor and record conditions of post accident reactor coolant gas valves and positions, containment sump and containment levels, Diverse Scram System, and auxiliary feedwater pumps and valves. Controls for the monitored systems are also provided.

Two rows of panels behind the main control board contain instruments and controls which require only occasional supervision and auxiliary equipment (e.g., power supplies) which must be convenient to other equipment in the control room but require no operator attention. The functions of these panels include component and raw water systems controls, auxiliary building ventilation and purge systems controls, fire detection and lighting.

Panels AI-106A and AI-106B are the control room air conditioning system control panels and are located out of the main operating area along the south wall of the control room. These panels contain controls and instrumentation for the control room air conditioning units, emergency filtration units, and the control room iodine monitor. These systems are designed to require minimal operator attention. These panels are part of the Engineered Safeguards.

7.6.3 Radioactive Waste Disposal System Control Panels

The radioactive waste disposal system is controlled by a combination of automatic, remote-manual, and local operations. Remote-manual operations are conducted from panel AI-100 located in the auxiliary building at elevation 989'-0" near the processing equipment. The panel also contain supervisory instrumentation for the automatic functions and annunciators.

Panel AI-100 is used to control the following functions:

a. Water transfer from the waste holdup tanks, hotel waste tanks, and spent regenerant tanks to the inlet treatment header or monitor tanks;

b. Treated water from the monitor tanks to the overboard header or for further treatment;

c. Gaseous waste release to the ventilation discharge duct.

In cases where redundant components are provided, selection of the equipment to be used is made from the panel. Equipment which is supervised from this panel, and normally operated automatically, can also be operated manually from this panel as follows:

1. Gas decay tanks;
2. Waste gas compressors;
3. Waste holdup recirculation pump;
4. Auxiliary building sump tank and pumps;
5. Spent regenerant tank pumps;
6. Waste filters (no controls);
7. Spent resin tank;
8. Containment sump pumps.

The portions of the liquid and solid radwaste processing system in the Radioactive Waste Processing Building are controlled from local panels near the mobile process equipment. Because of the nature of this equipment, each system has its own unique controls.

7.6.4 Miscellaneous Local Control Stations

As far as possible, the control of plant systems is concentrated in the control room. The auxiliary feedwater system can also be operated from an emergency panel located elsewhere. The diesel-generators can also be controlled from panels located in the diesel-generator rooms. As described previously the waste disposal system is controlled from outside the control room; other panels, described below, have also been found desirable to control local functions. The local panels are:

a. Auxiliary feedwater regulating panel (AI-179): This panel displays the water level in each of the steam generators and contains controls for the auxiliary feedwater valves and the turbine driven auxiliary feedwater pump and its associated recirculation control valve. The panel also contains a master transfer switch to transfer control of the auxiliary feedwater system from the control room to this point. These panels are part of the Engineered Safeguards.

b. Diesel-generator panels (AI-133A and AI-133B): These panels (see Section 7.3.6) include master switches which disconnect all external sources of supply from the station bus, shed all non-essential loads, disconnect all engine and generator protective devices except overspeed trip and isolate control from the safeguards panels, AI-30A and AI-30B. Operation of the master switch allows local operation of the diesel breaker emergency switch to connect the generator to the station bus. Each diesel- generator is also provided with an integral panel which permits starting the unit and isolating control of the engine from the control room. These panels are part of the Engineered Safeguards.

c. Temperature control panels for the auxiliary building ventilation systems: The panels contain controllers only (see Section 9.10.2.1); thermostats are located elsewhere.

d. Vacuum deaerator control panel (AI-105): This equipment normally operates automatically; the panel contains supervisory equipment.

e. Air compressor control panel: The air compressors run automatically; the panel includes unloading controls, supervisory instrumentation, and an operation sequence switch.

f. Condenser backwash panel (AI-123): This panel contains circulating water valve controls. An operator periodically reverses flow through each condenser shell in turn for a short period to dispose of debris in condenser tubes and at the tube inlets. The panel is in the turbine room at the basement level.

g. HP heater drain panel (AI-121): This panel contains level controllers for feedwater heaters 5 and 6. The equipment operates automatically. The panel is in the turbine room at the basement level.

h. LP heater drain panel (AI-122): This panel contains level controllers for feedwater heaters 1, 2, and 3. The equipment operates automatically. The panel is in the turbine room at the basement level.

i. Screen wash panel (AI-120): This panel contains screen controls, screen wash pump controls, circulating water pump discharge valve controls and supervisory instruments for this equipment. The circulating water pump and circulation pump discharge valve controls duplicate functions on the main control board.

j. Stator winding cooling and hydrogen control cabinet (AI-134): This panel includes annunciators and other instruments and controls for operating and monitoring the stator winding cooling water system. It also incorporates a flowmeter, pressure indicator, and other instruments, controls, and alarms for the generator hydrogen cooling system.

k. Alternate Shutdown Panels: Alternate Shutdown Panels (ASP) AI-185 and AI-212 have been installed next to the auxiliary feedwater panel AI-179 in the electrical penetration room at Elevation 1013'-0". These panels are part of the Engineered Safeguards.

The following instruments and controls are provided on this panel:

- Primary Loop Hot Leg Temperature Indicator
- Primary Loop Cold Leg Temperature Indicator
- Volume Control Tank Level Indicator
- Pressurizer Level Indicator
- Control Switch "Open-Close" with one set of indicating lights for HCV-239
- Open/Close indication for: HCV-240, PCV-103-1, PCV-103-2 and TCV-202
- Control Switch "Close-Trip" with one set of indicating lights for CH-1B
- Wide Range Neutron Flux Monitor

7.6.5 Features Which Enhance Safe Operation

All panels and consoles are enclosed. Panels within the control room are individually air cooled where required. All cables enter from the cable room below; the sleeves between the cable room and control room are sealed. A fire originating outside the control room could not spread to the control room through the duct system or cable sleeves. Further, since combustible materials are excluded from the control room as far as practical, the danger of fire in the control equipment is minimized.

Safety circuits have redundant channels. Two channels are generally provided for actuating circuits such as containment isolation valve closure signals. Four channels are provided for sensing circuits such as pressurizer level. Inside the panels, equipment in each channel is separated from adjacent channels by metallic separators to minimize the possibility that damage to one channel could cause damage to another channel serving the same purpose.

In some cases, such as the containment isolation valve panel, (AI-43A and AI-43B), two sections are provided, one for instruments and devices in each channel.

The panels are arranged to minimize the span of attention required from the operator so that he can better monitor plant conditions. Items most likely to require frequent or immediate attention are grouped toward the center of the panel and equipment requiring less frequent attention toward the ends of the panel. For example, reactivity control is close to the middle. The electrical distribution equipment is at one end. Items which are likely to need attention, at the same time or in succession, are grouped together as much as possible. For example, the steam dump valve controls which are used to stabilize reactor coolant parameters after plant heatup is completed are adjacent to the CEA controls which are used when the reactor is made critical. Such grouping minimizes the likelihood of operator error.

7.6.6 In-Plant Communication System

The in-plant communication system comprises two separate systems, a GAI-Tronics Transistorized Communication (GTC) system and a sound-powered, telephone system. The dial telephone system is not a plant-wide system and is not intended for in-plant communication. The dial telephones each have individual numbers and are connected to the District's internal telephone system and the local telephone company exchange.

OPPD's 800 Mhz radio communication trunking system has been expanded to Fort Calhoun and is available to various groups at the plant. This system is not considered a plant wide system.

The basic GTC system is made up of handsets which have a local speaker amplifier and are distributed at strategic locations throughout the plant. Jack stations are also located in containment for use with portable handsets and amplifiers.

Two single channel sub-systems complement the basic three channel system. As an aid to fuel handling operations, handsets at the fuel pool area and the refueling crane inside the containment permit intercommunication between these areas and the control room. Certain handsets in the control room are, therefore, equipped with an additional channel. The GTC system is supplied from the 120-Volt instrument ac system (see Section 8.3.5).

The sound-powered telephone system is provided to facilitate maintenance and to back up the main electronic system. Sound-powered telephone jacks are distributed throughout the plant at certain GTC system locations. Portable handsets and portable headsets are provided. The system has two channels to provide the facility for two separate conversations utilizing up to the total number of handsets and headsets available.

7.6.7 Off-site Communication

Off-site communication comprises five separate systems, a dial telephone system connected to the District's internal telephone system and the local telephone company in Blair and Omaha; dedicated leased telephone lines for various functions; a microwave transmission system; 800 Mhz two way radio system and backup radio systems (Washington County Sheriff contact and emergency 800 Mhz radio communication backup with all OPPD's power plants) provided in case of failure of telephone and/or radio system.

7.6.8 Alternate Shutdown Capability

Alternate shutdown capability is provided for use in the event of an accident situation which renders the control room uninhabitable. Located in the auxiliary building the alternate shutdown panel contains the necessary instrumentation and control equipment to allow the operator to safely bring the plant to hot shutdown status and maintain that status until sufficient corrective measures can be taken to allow and maintain a cold shutdown.

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8.2 NETWORK INTERCONNECTIONS

The station output is supplied to the 345-kV transmission system through the substation located at the plant site. Figure 8.2-1 is a one-line diagram of the transmission system in the vicinity of the plant site.

8.2.1 Distribution of Station Output

The generator output is fed through a two-winding 22-345-kV delta-wye transformer to a breaker and a half scheme in the switchyard. Three 345 kV transmission lines connect the on-site switchyard to Omaha; Lincoln, via the Lincoln Electric System (LES) Wagener Substation; and Sioux City, via the Mid-American Energy, Inc. Raun Substation. Each of the three 345-kV lines connected to the on-site substation has sufficient capacity to carry the station output.

The plant substation is arranged as a breaker and a half scheme and includes high speed relaying for line and bus protection. Primary relaying protection of the transmission lines is accomplished via power line carrier relaying. Secondary relaying using telephone pilot wire relaying was also installed to ensure protection.

Line design exceeds the requirements of National Electric Safety Code for heavy loading district, grade B construction. The line design was based on calculated lightning performance of less than one outage per hundred miles per year.

A detailed stability analysis involving the Fort Calhoun Power Station for the summer of 1971 conditions was conducted by OPPD in 1968. In representing the 345 kV system associated with the Fort Calhoun unit, the major portion of the bulk transmission system in seventeen states was included. In order that the system investigation would represent the most severe conditions, the maximum tolerable power transfer (1100 megawatts south to north) was represented in the Mid-Continent Area Power Pool (MAPP) (formerly Mid-Continent Area Power Planners) system as a predisturbance condition.

The study was conducted using the Westinghouse Dynamic Stability Program. The major generating units in the MAPP area were represented in detail complete with generator electrical constants, turbine-generator inertia constants, excitation system description and electrical constants, and speed-governor description and control constants.

Since previous studies had indicated that the system may not go unstable until after the second or third oscillation, a five second study duration was utilized to ensure that the most severe oscillation was included.

Both three-phase and single line to ground faults were examined as follows:

1. Primary clearing with reclosing into a permanent fault.

2. Breaker failed to operate at the fault location and the fault is finally cleared by back-up operation tripping an additional line.

3. The Fort Calhoun unit was tripped by a back-up relay.

Conclusions from this study indicate that a three-phase fault at the Fort Calhoun substation with a stuck breaker at Fort Calhoun which allows the unit to feed the fault for thirteen cycles and is then cleared by back-up relaying will not cause the remainder of the system to become unstable.

The effect of a trip of the largest unit in the MAPP system at the time, the 800 MWe NPPD Cooper Station Unit No. 1, was investigated by Stone & Webster, consulting engineers, in 1967. The investigation considered disturbances during 1000 megawatt power transfers both north and south. Conclusions indicated the system to be stable for all conditions of load transfer.

Subsequent studies by the MAPP Reliability Studies Task Force have simulated various conditions for the 1982 and 1986 summer seasons. Although these were not thorough studies of the area around Fort Calhoun, they were intended to locate the most severe "probable and extreme" disturbances, as defined by MAPP, within the power pool. None of the disturbances simulated resulted in a complete collapse of the system, causing loss of station power at the Fort Calhoun Station.

The results of these stability studies indicate that neither the loss of the Fort Calhoun Station unit nor the loss of the largest unit in the MAPP system will adversely affect the remainder of the transmission system. The two sources of off-site power, the 161 kV line and the 345 kV system, will remain intact.

The 345 kV and the 161 kV lines entering the switchyard from the north and west are widely separated except for two 345 kV/161 kV crossings west of the switchyard (see Figure 8.2-2).

The 345-kV substation switching power (breaker control and operation) is provided by one of two batteries.

Each battery is connected to a battery charger fed from the switchyard 13.8 KV distribution system in normal and emergency operation. This is the normal layout for all OPPD transmission and distribution substation service.

The 161 KV substation uses two batteries in an arrangement similar to the 345 KV substation where both the normal and emergency battery charger power feed are from the switchyard 13.8 KV distribution system.

The 345 kV and 161 kV lines from the switchyard to the plant were located as far apart as possible. Those points of minimum separation, including one crossing, were established by the location of the transformers, and by the orientation of equipment in the switchyard.

In all, 345 kV line 3423 crosses 161 kV line 1587 at one location. OPPD records show that of fifty-five crossings of transmission lines, i.e., 345/161 kV and 161/69 kV, in operation at the time of the study, no major line failures had occurred at the points of crossing. The fact that the 345 kV towers have been designed to withstand 1.7 times the wind loads of our conventional 161 kV double circuit steel tower line, plus a special one inch radial ice loading design, further demonstrates the improbability of a failure at one of the 345 kV/161 kV crossings at the Fort Calhoun site. Should a failure of either a conductor or shield-wire on the 345 kV line occur the 161 kV line breakers would trip, thus preventing damage to electrical equipment. It is considered unlikely that the 161 kV line would be physically damaged by the falling of any or all shield-wires and conductors of the 345 kV line at a crossing point.

8.2.2 Station Service Power Supply

Power for the 4.16-kV station auxiliary system is available from two separate systems, either of which has sufficient capacity for all of the auxiliaries. One source is the generator 22-kV bus, tapped between the generator disconnect switch, DS-T1, and the 22-345 kV generator transformer; the other source is the 161 kV system which is normally connected to both the OPPD generating system in Omaha and generating system in Sioux City. Each of these two sources feeds the station auxiliaries through two transformers.

Backup auxiliary power is provided by backfeeding from the 345 kV system via the main transformer and the 22-4.16 kV auxiliary transformers following a generator trip and manual opening of generator disconnect switch DS-T1. DS-T1 current interrupting capacity limitations require that interlocks be provided to prevent switch opening unless the 345 kV generator circuit breakers have tripped, generator excitation has been removed, and the generator bus is deenergized, the isolated phase bus forced air cooling system has been shut down, and the four 4.16-kV bus breakers associated with the 22-kV supply are open.

The disconnect switch DS-T1 operator is actuated exclusively with a key operated manual control switch.

To further ensure switch reliability, DS-T1 is normally exercised following each generator trip or generator shutdown, and the contacts are normally inspected before the switch is reclosed.

The adequacy of the switch design was demonstrated by magnetizing current interruption, voltage impulse, current impulse, and heat run factory tests.

Should loss of offsite power occur during a DBA, bus undervoltage relays together with auxiliary relays will operate to:

a. Trip Engineered Safeguards and trip other, nonessential loads.

b. Bring the Diesels from "Idling" to Full Speed and Voltage.

c. Reset the four Engineered Safeguards Sequencers.

Diesel Generator Breaker closure on buses 1A3 or 1A4 will cause the associated timers on the Sequencers to begin to time out and restart the first group of safeguards loads.

Protective relays will operate through lockout relays to disconnect buses 1A3 and 1A4 from the faulted offsite power source. With no voltage on buses 1A3 and 1A4, completion of 4.16-kV load shed, and proper voltage and speed at the diesels, the diesel breakers will close automatically to restore power to the buses.

Normal protective trip devices for the diesel engine and generator are bypassed except for the overspeed trip.

A 13.8 kV emergency power system is available from the 13.8 kV Distribution system to allow for plant shutdown in the event that the normal plant power supply, including the emergency diesel generators, is lost.

8.5 INITIAL CABLE INSTALLATION DESIGN CRITERIA

The following summarizes the cable installation design criteria intended to preserve the independence of redundant Reactor Protective systems and of those systems designed as Engineered Safeguards. The Cable and Conduit Schedule Notes, Figure 8.5-1, provides the standard design criteria for cables and conduits. Deviation from the standard criteria is acceptable provided an analysis has been completed which justified the deviation.

8.5.1 Cable Separation Criteria

Cable separation criteria of redundant Reactor Protective and Engineered Safeguards circuits are as follows:

a. Redundant Reactor Protective and Engineered Safeguard circuits are routed from their sensors to the cable room in separate cable trays, conduits, containment penetrations and junction boxes.

b. Cables are identified according to the notes shown in Figure 8.5-1. These notes ensure segregation of redundant circuits with special emphasis placed on Reactor Protective and Engineered Safeguard circuits. See notes #2, 5, 7, and 15 of Figure 8.5-1.

c. Redundant Reactor Protective and Engineered Safeguard instruments are identified by tag numbers prefixed A, B, C, or D followed by a slash (/) in agreement with the cable prefix.

d. The auxiliary and containment building cable trays are divided into four basic systems. These systems are identified on the drawings as EA, EB, EC, and ED. These designations agree with the cable numbering system as stated in Figure 8.5-1.

   Two tray systems are assigned to each floor with a minimum horizontal separation of 2-3". Where this minimum horizontal separation cannot be maintained suitable metallic barriers are installed.

The cable tray systems are assigned as follows:

Location	Floor Elevation	Tray Systems
Auxiliary Building	971'	EA, EB
Auxiliary Building	989'	EA, EB
Auxiliary Building	1007' to 1013'	EC, ED
Auxiliary Building	1025'	EA, EB
Auxiliary Building	1036'	EC, ED
Containment Building	994'	A - North half EB - South half
Containment Building	1013'	C - South half ED - North half
Containment Building	1045'	o cable trays

Cables to Reactor Protective and Engineered Safeguard equipment whose prefix differs from that of the nearest tray system are routed separately to the matching tray system.

e. The cable spreading room contains the cable tray, conduit, and junction box system for the routing of cables to the control boards, auxiliary instrument panels and the ERF computer. In addition, the cable spreading room contains items to support the operation of the plant including such items as fire extinguishers, fire detectors and panels, fire suppression (Halon) for both room 70 and control room walk-in cabinet, emergency lighting and Gaitronics communication, control room sanitary drain, control room air conditioner room floor drains, mechanical equipment drain line and isolation valve, ventilation dampers, 800 mhz radio system junction box JB-622a, electrical power receptacles, condenser vacuum gauge lines, cables abandoned in place, control room delta pressure sensor, and lighting including panel LP-7.

The cable tray and conduit arrangement is shown on Figure 8.5-2. All trays are run in vertical banks with a minimum vertical separation of 12 inches.

Engineered Safeguard cables are separated by metallic barriers right up to the control boards in accordance with note #7 of Figure 8.5-1.

f. The E prefixed cables inside the screenhouse and between the plant building and screenhouse are routed in separate conduits, tray sections, or in separate duct bank conduits (plastic tubes embedded in concrete).

The pull box and manhole layout is as follows:

1. Pull boxes

   There are two pull boxes along the outside of the south auxiliary building wall. The pull boxes are divided in sections by asbestos-cement compound plates. One pull box contains EA and EC cables in separate sections, and the other pull box contains EB and ED cables, also in separate sections. In conformance with Note 22 of Figure 8.5-1, a metallic barrier is placed inside each section containing E prefixed cables in order to segregate them from the non-E prefixed cables.

2. Manholes

   There is one manhole between the pull boxes and the screen house. The cables are in cable trays and the routing is in conformance with the Cable and Conduit Schedule Notes (Figure 8.5-1). There is a 6" minimum thickness concrete wall separating cable trays with EA and EC cables from cable trays holding EB and ED cables.

   The cable tray system in the cable room does not have covers since the installation consists mainly of instrument and control cables, basically 120V AC, 125V DC, or low energy signal and computer control circuits.

   There are no 4160V or 480V power cables installed in cable trays in the cable spreading room.

g. The criteria governing the separation of power cables from those used for control and instrumentation are stated in notes #10, 11, 12 and 13 of Figure 8.5-1. In general these cables are grouped in separate trays and the notes in Figure 8.5-1 apply to those areas where physical limitations, etc., preclude this.

h. The intermixing of non-vital cables with Reactor Protective or Engineered Safeguard cables is prohibited by note #22 of Figure 8.5-1.

i. The cable for each redundant sensor of a protection channel is assigned a different prefix (EA, EB, EC, or ED) and is routed separately in accordance with the notes on Figure 8.5-1. If cables of two different protection channels are located in the same area, note #7 of Figure 8.5-1 ensures that only those cables with the same prefix will be grouped together.

8.5.2 Cable Protection Against Missiles

The cable installation design criteria described in USAR Sections 8.5.1, 8.5.3, and 8.5.5 along with the following summary comprise the methods used to ensure that in areas containing high pressure piping or where mechanical damage is possible, such as from missiles generated by rotating equipment, no single credible incident could damage more than one cable raceway of a redundant system. Redundant circuits referred to in this summary are those associated with Reactor Protective and Engineered Safeguards systems.

The methods employed for the protection of these cables are as follows:

a. Each redundant channel is routed in a separate cable tray system.

b. Each redundant channel is routed in separate conduits and junction boxes.

c. Each redundant channel is routed through a separate containment penetration.

d. Only two out of the four redundant cable tray systems are assigned to each floor elevation and these are horizontally separated.

e. Cable trays are installed with covers.

f. In the auxiliary building main cable tray systems are located in corridors where the amount of rotating equipment is at a minimum.

g. In the containment cable trays are located in protected areas (see USAR Section 5.8.1). When a redundant electrical component is located inside the shield walls surrounding the reactor coolant loops, conduit is used.

h. Instrument channels initiating Reactor Protective or Engineered Safeguard circuits employ two-out-of-four matrices. These circuits will still function even with the loss of two of their separately routed input channels.

8.5.3 Electrical Penetration Separation Criteria for the Containment Building

The separation criteria for the electrical penetrations in the containment building are as follows:

Electrical penetrations are of the canister type. A description of the canister is included in the USAR Section 5.9.3. An arrangement drawing showing the spacing and service of each canister is shown on Figure 5.9-18. This figure shows that separate canisters are assigned for the following class cables and is in accordance with Figure 8.5-1. The platform and floor shown on Figure 5.9-18 exists on both sides of the containment wall.

4160V AC Power A
4160V AC Power B
4160V AC Power C
4160V AC Power D
Power EA (480V, 120/208V and 125 DC)
Power EB (480V, 120/208V and 125 DC)
Power EC (480V, 120/208V and 125 DC)
Power ED (480V, 120/208V and 125 DC)
Power A&C (480V, 120/208V and 125 DC)
Power B&D (480V, 120/208V and 125 DC)
Control EA (125V DC & 120V AC)
Control EB (125V DC & 120V AC)
Control EC (125V DC & 120V AC)
Control ED (125V DC & 120V AC)
Control A&C (125V DC & 120V AC)
Instrumentation EA
Instrumentation EB
Instrumentation EC
Control B&D (125V DC & 120V AC)
Instrumentation EA
Instrumentation EB
Instrumentation EC
Instrumentation ED
Instrumentation A&C
Instrumentation B&D
Coax
EA
Coax EB
Coax EC
Coax ED

Reactor Protective and Engineered Safeguard cables with different prefixes are routed through separate canisters. Cables are routed from the canister in separate tray or conduit in accordance with the cable separation criteria stated in USAR Section 8.5.1.

8.5.4 Cables

Cable criteria are as follows:

a. Cable tray loading is in accordance with notes #17, 18, and 20 of Figure 8.5-1. Note #17 applies to 5 kV power cable only. Note #20 is further described as follows:

   1. <u>600 Volt Class E Prefixed Power Cable</u>

      The fill in cable trays shall generally not exceed 40 percent of the rectangular area derived from the height of the cable tray side times the cable tray width. Fill exceeding 40 percent shall be justified by analysis.

      The 40 percent fill is defined as the sum of the cross-sectional areas of all cables in the tray. For triplexed cable only, the cross-sectional area includes the spaces between the three conductors as enclosed in an encompassing circle.

   2. <u>600 Volt Class Non-E Prefixed Power Cable</u>

      In general the same criteria as for E prefixed cable shall apply.

   3. <u>600 Volt Class E and Non-E Prefixed Control Cable Used For 125 Volt DC and 125 Volt AC Control Circuits</u>

      The fills in cable trays shall generally not exceed a maximum of 50 percent of the rectangular area derived from the height of the cable tray side times the cable tray width. Fill exceeding 50 percent shall be justified by analysis.

      The 50 percent fill is defined as the sum of the cross-sectional areas of all cables in the tray.

b. Cable environmental qualification is described in USAR Section 1.4.8.2 and 1.6.16.

c. Cable splicing in cable trays is used only for connection of incoming and outgoing cables with containment electrical penetration conductors.

d. Cable derating is in accordance with established methods as described in IEEE Publication No. S-131-1 - IPCEA (currently ICEA) Publication No. P-46-426, titled "Power Cable Ampacities - Vol. I - Copper Conductors".

   An ambient temperature of 50° C. is assumed for exposed conduit and cable tray and an ambient temperature of 20° C. is assumed for underground ducts. The maximum allowable continuous conductor temperature is 85° C.

e. As part of the electrical system's design, power cables have been oversized to ensure equipment operability. In general, a 125% full load current criteria per National Electric Code Article 430-22 and derating procedures outlined in "e" above were used.

f. Cable and wireway markings

   1. Generally, whenever a conduit enters or leaves a box or tray, it is marked on each side of the box or at the tray, in accordance with the Cable and Conduit Schedule, with the identifying number of the cable or cables in the conduit run to which they are attached.

   2. Cables are identified with suitable markers in accordance with the identifying number assigned in the Cable and Conduit Schedule.

Engineered Safeguards cables are a subset of Safety Related cables which are identified by an "E" prefix and are separated for easy identification by their distinctive colored jacket or jacket banded with colored tape every three feet as listed below:

Cable Number Prefix	Jacket Color
EA	Red
EB	Green
EC	Yellow
ED	Blue

3. Wires are identified by individual wire numbers or letters at both ends of wires and at terminal boards. Wire identification corresponds to that shown on the elementary and connection diagrams.

8.5.5 Fire Protection Requirements for Cables

a. Flame resistance qualifications

The cables that are installed in trays must meet the flame resistance qualification test described in the USAR Section 1.4.8.2.

b. Temperature monitoring of cables

The conservative approach of cable tray loading and derating procedures, outlined in USAR Section 8.5.4, precludes the necessity of monitoring cable temperatures in trays.

c. Fire detection system, connected to electrical bells and annunciation in the control room is described in USAR, Section 9.11.2.1.

Location of detectors at strategic positions is in accordance with Figures 8.5-4 through 8.5-11.

d. Fire stops

There are fire stops for slots and openings in walls and floors through which cable trays pass.

8.5.6 Process Instrumentation Inside Containment Building

The criteria for the process instrumentation inside the Containment Building were as follows:

a. Process instruments within the containment are located in shielded areas accessible for maintenance.

   Redundant instruments for safety instrumentation are identified by tag numbers prefixed by a capital letter A, B, C, or D followed by a slash(/). Sensing lines to these redundant instruments are run from separate sensing points. Redundant instruments within the containment for a safety channel are located on physically separate racks or on a common rack. However, where these instruments are located on a common rack metal barrier plates are provided to maintain separation between all A/, B/, C/, and D/ instruments and lines. Redundant instrument racks were not placed closer than three feet from each other unless they were separated by a wall or furnished with a metallic plate on their sides.

   Redundant instrument sensing lines were not placed closer than three feet from each other unless they are separated by an adequate shield (steel plate, steel channel, concrete wall, etc.) to protect the lines against mechanical injury. In the case where two redundant sensing lines cross each other the mechanical separation was provided for a radius of at least two feet from the point of crossing.

b. All cable trays in the containment containing redundant instrumentation leads are located in a protected area. This area is outside the concrete shield walls surrounding the reactor coolant loops (see USAR Section 5.8.1). When thermocouples or RTD's are located inside these walls, their cables are routed in conduit.

Cables of redundant instruments are identified in accordance with Figure 8.5-1. To maintain separation, these cables are generally routed in the following manner to the penetration area.

Cables prefixed EA----------	Routed counterclockwise above floor elevation 994'
Cables prefixed EB----------	Routed clockwise above floor elevation 994'
Cables prefixed EC----------	Routed clockwise above floor elevation 1013'
Cables prefixed ED----------	Routed counterclockwise above floor elevation 1013'

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

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

9.2.1 Design Bases

The chemical and volume control system (CVCS) is designed to perform the following functions:

a. Maintain the reactor coolant chemistry and purity within specifications.

b. Maintain the reactor coolant volume within programmed limits.

c. Provide a means of adding and removing boron to control reactor reactivity level changes.

d. Provide a storage location and makeup water source to compensate for reactor system volume changes during plant heat up and cooldown at the maximum allowed rate.

e. Provide a storage location and makeup water source to compensate for reactor system volume changes due to power level changes.

f. Provide a means to hydrostatically test the reactor coolant system.

g. Provide a means for hot leg injection in the post-LOCA long term cooling.

h. Provide a means for collecting and reusing reactor coolant pump mechanical seal controlled leakage.

i. Provide a system for mixing and storing concentrated boric acid solution for use in the reactor coolant system.

j. Provide a means for addition of chemicals to the reactor coolant system, e.g. pH control in the form of Li_7OH; oxygen control in the form of N_2H_4 when the plant is cold; hydrogen peroxide for forced oxygenation during some plant shutdowns.

k. Provide a means of controlling reactor system hydrogen concentration.

l. Provide auxiliary spray flow to the pressurizer.

For more supporting details of the Chemical and Volume Control System (CVCS) components, the Technical Specifications, the CQE Manual, the EEQ Manual, and the Regulatory Guide 1.97 Responses should be consulted.

9.2.1.1 Design Cyclic Loads

The following design cyclic transients, which include conservative estimates of the operational requirements for the components listed in Section 9.2.1.2, were used in the fatigue analysis required by the applicable design codes for certain components within the CVCS:

a) 500 reactor heatup and cooldown cycles, at a heating and cooling rate of 100°F/hr, during the system 40-year design life.

b) 500 reactor power change cycles over the range of 10 percent to 100 percent of full load with a ramp load change of 10 percent of full load per minute increasing or decreasing.

c) 500 reactor cycles of 10 percent of full load step power changes increasing from 10 percent to 90 percent of full power and decreasing from 100 percent to 20 percent of full power.

d) 10 cycles of hydrostatic testing the reactor coolant system at 3125 psia and at a temperature at least 60°F above the Nil Ductility Transition Temperature (NDTT) of the limiting reactor component.

e) 200 cycles of leak testing the reactor coolant system at 2100 psia and at a temperature at least 60°F greater than the NDTT of the reactor vessel.

f) 400 reactor trips when at 100 percent power.

g) 1000 cycles of Maximum Purification

h) 8000 cycles of Boron Dilution

In addition to the above list of normal design transients the following abnormal transients were also considered when arriving at a satisfactory usage factor:

i) 40 cycles of loss of turbine load with delayed reactor trip from 100 percent power.

j) 40 cycles of total loss of reactor coolant flow when at 100 percent power.

k) 5 cycles of loss of secondary system pressure.

l) 80 cycles of Low Volume Control Tank Level.

m) 500 cycles of Loss of Charging.

n) 700 cycles of Loss of Letdown.

o) 200 cycles of Long Term Letdown Isolation (in excess of 1 hour).

p) 700 cycles of Short Term Letdown Isolation (up to 1 hour).

q) 200 cycles of Intermittent Manual Charging (significant only for charging nozzles).

The number of cycles, defined for each transient above, are not design limits but rather engineering estimates as to the number of cycles that might be incurred over the assumed design life of the plant. The significance of these numbers is that the cumulative fatigue usage of all transients, as defined above, is no greater than 1.0 which is the design limit imposed by the applicable design codes. An assumed number of cycles for each transient is necessary to calculate a fatigue usage contribution for that transient. An increase in any one of the cycle number estimates, above, may necessitate a decrease in one or more of the others to satisfy the cumulative usage limit of 1.0. Although not design limits, the defined cycles are useful benchmarks against which to compare the actual incurred cycles which are required to be recorded by Technical Specification 5.10.2.f as implemented by SO-O-23. In the event that the incurred cycles are likely to or actually exceed these estimates, the supporting calculations may require revision to justify changes in these numbers.

9.2.1.2 Design Service Life Considerations

The major CVCS system components were designed for a 40-year service life (which is the basis for the licensed number of operating years for the plant). This number is not a design limit but rather an assumed engineering basis for estimating the number of cycles for transients which significantly contribute to the fatigue usage of the affected components. The design limit is that the cumulative fatigue usage, for all transients over the life of the plant, must be no greater than 1.0.

CVCS Components Designed for Fatigue	Code Used for Fatigue Design
Charging Nozzles	ASME Section III
Regenerative Heat Exchanger	ASME Section III, Class A*
Letdown Heat Exchanger	ASME Section III, Class A*
Piping	USAS B31.7, Class 1

***NOTE**: The fatigue rules of Class A were conservatively applied even though the vessels were constructed as Class C which does not require fatigue analysis.

Thermal sleeves were used in the charging nozzles, regenerative heat exchanger tube side inlet, tube side outlet, shell side outlet and interconnecting pipe nozzles and letdown heat exchanger tube side inlet nozzle to minimize the fatigue usage, associated with fluid thermal transients. The thermal sleeves are not part of the code pressure boundary and are therefore not subject to the fatigue limitations imposed by the codes applicable to the nozzles. The design of the sleeves is such that their expected fatigue life is in excess of the assumed design life of the plant.

9.2.2 System Description

9.2.2.1 General

The chemical and volume control system is shown in the simplified block diagram, Figure 9.2-1, and the detailed piping and instrumentation diagrams, P&ID E23866-210-120 and E-23866-210-121. Coolant normally flows through the chemical and volume control system, as shown by the heavy lines in P&ID E23866-210-120 and E-23866-210-121. The letdown coolant from the cold leg of the reactor coolant system passes through the tube side of the regenerative heat exchanger and is partially cooled. The fluid pressure is reduced and the flow rate is regulated by the letdown control valves. The temperature is further reduced in the letdown heat exchanger to a level consistent with long ion exchanger resin life. The pressure in the system between the letdown control valves and the letdown heat exchanger is controlled by the letdown back pressure control valve to prevent flashing. The coolant then passes through an ion exchanger and a filter and is sprayed into the volume control tank. The charging pumps take suction from the volume control tank and return the coolant to the reactor coolant system by way of the shell side of the regenerative heat exchanger. The heat exchanger transfers heat from the letdown coolant to the charging coolant before the charging coolant is returned to the reactor coolant system.

If the valve control switches are selected to AUTO, and the level in the volume control tank reaches the high level setpoint, the letdown flow is automatically diverted to the waste disposal system. If the level in the volume control tank reaches the low-low level setpoint, makeup water from the safety injection and refueling water tank (SIRWT) is automatically supplied to the suction of the charging pumps.

With the level in the normal operating level band, the volume control tank has sufficient capacity to accommodate a full to zero power decrease without makeup system operation.

The boric acid concentration and chemistry of the coolant are maintained by the chemical and volume control system. Concentrated boric acid solution is prepared in a batching tank and is stored in two concentrated boric acid storage tanks. Two pumps are provided to transfer concentrated boric acid to the volume control tank or the charging pump suction. The piping is arranged such that the boric acid may be mixed with demineralized water in a predetermined ratio. The solution is introduced to the reactor coolant system by the charging pumps.

Chemicals are introduced to the reactor coolant system by means of the boric acid pump which removes the chemical solution from the concentrated boric acid tanks or by means of a metering pump which removes the chemical solution from a chemical addition tank and transfers it to the charging pump suction header.

The reactor coolant system may be tested for leaks when the plant is shutdown using a charging pump for pressurization. The system is also provided with connections for installing a hydrostatic test pump.

Any substantial leakage in the reactor coolant system may be detected while the plant is at power by monitoring pressurizer level, volume control tank level, and charging and letdown rates.

The general performance parameters are given in Table 9.2-1.

Table 9.2-1 - "General Performance Parameters"

Nominal Letdown Flow, gpm	36
Nominal Purification Flow Rate, gpm	36
Nominal Charging Flow, gpm	40
Reactor Coolant Pump Controlled Bleed-off (4 pumps), gpm	4
Maximum Letdown Temperature at Loop, °F	547
Nominal Charging Temperature at Loop, °F	440
Ion Exchanger Operating Temperature, °F	120

9.2.2.2 Volume Control

The CVCS automatically adjusts the volume of water in the reactor coolant system using a signal from the level instrumentation located on the pressurizer. The system reduces the amount of fluid that must be transferred between the reactor coolant system and the CVCS during power changes by employing a programmed pressurizer level set point which varies with reactor power level. The set point varies linearly with the average reactor coolant temperature measured across a steam generator. This linear relationship is shown in Figure 4.3-10. The control system compares the programmed level set point with the measured pressurizer water level. The resulting error signal is used to control the operation of the charging pumps and one letdown valve, as described below. The pressurizer level control program is shown in Figure 4.3-11.

The pressurizer level control program regulates the letdown flow by adjusting the letdown control valve, so that the reactor coolant pump controlled bleed-off plus the letdown flow matches the input from the operating charging pump. When the equilibrium is disturbed by a power change or for any other reason, a decrease in level starts one or both nonoperating charging pumps to restore level, and an increase in level increases the letdown flow rate and initiates a backup signal to stop the two standby charging pumps.

The volume control tank coolant level may be automatically controlled, but is normally manually controlled by Operations using feed and/or bleed. When the level in the tank reaches the high-level setpoint, the letdown flow may be automatically diverted to the waste disposal system. If automatic operation is selected and the level in the tank reaches the low-level setpoint, a preset blend of concentrated boric acid and demineralized water can be introduced into the volume control tank. Should the level in the tank reach the low-low level setpoint, the system automatically closes the outlet valve on the tank and switches the suction of the charging pumps to the safety injection and refueling water tank (SIRWT).

The volume control tank can store enough coolant within its normal operating level to compensate for a full to zero power decrease in the reactor coolant volume without requiring makeup. The tank is supplied with hydrogen and nitrogen gas. Gases are vented to the waste gas vent header.

9.2.2.3 Chemical Control

The CVCS purifies and conditions the coolant by means of ion exchangers, filters, degasification and chemical additives. The purification ion exchangers contain a mixed bed resin which removes soluble nuclides by ion exchange and insoluble particles by the filter action of the resin beds.

A cation ion exchanger is provided for the removal of cesium and/or lithium from the coolant if required.

Cartridge-type filters located downstream of the ion exchangers retain resin fines and remove insoluble particles that may pass through the resin bed.

Dissolved gases may be removed from the coolant by venting the volume control tank and purging with nitrogen as required.

The reactor coolant is chemically conditioned to the typical conditions shown in Table 9.2-2 by:

a. Hydrazine scavenging to remove oxygen during startup;

b. Maintaining excess hydrogen concentration to control oxygen concentration during operation;

c. Chemical additives to control pH during operation.

d. Hydrogen peroxide for forced oxygenation during some plant shutdowns. The following restrictions apply to addition of hydrogen peroxide:

   1) On Shutdown Cooling, RCS temperature <180°F, Pressurizer liquid temperature <250°F.

   2) RCS hydrogen $\leq$5 cc/kg.

3) I-131 ≤0.025 µCi/gm.

4) Hydrogen peroxide used: ≤30%, unstabilized.

5) Hydrogen peroxide added via CVCS Chemical Addition Tank.

6) RCS hydrogen peroxide residual 2.5 to 10 ppm.

7) Bypass CVCS Ion Exchangers prior to addition.

The chemical addition tank and metering pump are used to feed chemicals to the charging pumps which inject the additives into the reactor coolant system. The concentration of hydrogen in the reactor coolant is controlled by maintaining a hydrogen overpressure in the volume control tank. Hydrogen gas is maintained in the volume control tank when lithium hydroxide is used for pH control.

The chemical and volume control system is designed to prevent the activity of the reactor coolant from exceeding approximately 235 µCi/cc with 1 percent failed fuel elements.

When adding hydrogen peroxide there is a possibility of an explosive environment being created in tanks that are connected to the RCS. Thus the procedure controlling addition of hydrogen peroxide requires sampling and analysis of the Quench Tank, VCT, Vent Header, in service Waste Gas Decay Tank and the Reactor Coolant Drain Tank for an explosive environment before and after the addition of hydrogen peroxide.

Table 9.2-2 - "Typical Reactor Coolant and Deaerated Primary Makeup Water Chemistry"

	Reactor Coolant	Deaerated Primary Makeup Water
Specific Conductivity, Prior to Additives, micromhos/cm (maximum)	40	2.0
Nominal pH (77°F), range	4.5 to 10.2	6.0 to 8.0
Hydrogen (77°F), range, cc(STP)/Kg	27-50	-
Oxygen (77°F), maximum (ppm)	0.1	0.1
Halogens		
Chlorides, ppm (maximum)	0.15	0.15
Fluorides, ppm (maximum)	0.10	0.10
LiOH, ppm	LiOH Program (Ref. 9.2-1)	-
Suspended Solids, maximum, ppm	0.010 for Plant Modes 1 & 2	
Boric Acid Concentration		
Maximum (77°F), ppm	15,000	-
Nominal (77°F), ppm	0 - 2,500	-

9.2.2.4 Reactivity Control

The boron concentration of the reactor coolant is controlled by the CVCS to:

a. Optimize the position of the control element assemblies;

b. Compensate for reactivity changes caused by reactor coolant temperature variations, core burnup and xenon concentration variations;

c. Provide a margin of shutdown reactivity for maintenance, refueling or emergencies.

The system includes a batching tank for preparing the boric acid solution, two tanks for storing the solution, and two pumps for supplying boric acid solution to the makeup system.

Normally, the boric acid concentration of the coolant is adjusted by feed and bleed. To change concentration, the makeup (feed) system supplies either demineralized water or concentrated boric acid to the volume control tank and the letdown (bleed) stream is diverted to the waste disposal system. Toward the end of a core cycle, the quantities of waste produced due to feed and bleed operations becomes excessive and one of the two deborating ion exchangers is then used to reduce the boron concentration.

The system adds boric acid to the reactor coolant and thereby decreases reactivity at a sufficient rate to override the maximum increase in reactivity due to cooldown and the decay of xenon in the reactor.

The control element assemblies (CEAs) can decrease reactivity far more rapidly than the boron removal system can increase reactivity.

The charging pumps may be used to leak test the reactor coolant system at normal operating pressure when the plant is shutdown. Leaks in the reactor coolant system may be detected while the plant is at power by monitoring pressurizer level, volume control tank level, and charging and letdown rates.

9.2.3 System Components

The major components of the chemical and volume control system and their functions are described in this section.

9.2.3.1 Regenerative Heat Exchanger

The regenerative heat exchanger transfers heat from the letdown stream to the charging stream. Materials of construction were primarily austenitic stainless steel. The characteristics of the regenerative heat exchanger are given in Table 9.2-3.

Table 9.2-3 - "Regenerative Heat Exchanger"

Item No.	CH-6
Quantity	1
Type	Double Shell and Tube, Horizontal
Code	ASME III, Class C, 1968
Tube Side (Letdown)	
Fluid	Reactor Coolant
Design Pressure, psig	2485
Design Temperature, °F	650
Materials	Austenitic Stainless Steel
Pressure Loss at 36 gpm, psi	6
Pressure Loss at 116 gpm, psi	40
Shell Side (Charging)	
Fluid	Reactor Coolant
Design Pressure, psig	3025
Design Temperature, °F	650
Materials	Austenitic Stainless Steel
Pressure Loss at 40 gpm, psi	4
Pressure Loss at 120 gpm, psi	50

Operating Parameters

Tube Side (Letdown)	Normal	Unbalanced Charging With Heat Transfer	Maximum Purification	Unbalanced Letdown
Flow, gpm	36	26	116	116
Inlet Temperature, °F	547	547	547	547
Outlet Temperature, °F	207	167	375	421
Shell Side (Charging)				
Flow, gpm	40	120	120	40
Inlet Temperature, °F	120	120	120	120
Outlet Temperature, °F	440	208	303	520
Heat Transfer, Btu/hr	6.48×10^6	5.19×10^6	15.30×10^6	8.26×10^6

The following explanation is presented to explain the reason for changing the classification of the regenerative heat exchanger from an ASME Section III, Class A, 1968 vessel (PSAR page IX-2-8) to an ASME Section III, Class C vessel in Table 9.2-3 above: The Fort Calhoun Station regenerative heat exchanger was originally required to be an ASME Section III, Class A vessel following a similar classification assigned to the Palisades regenerative heat exchanger. A Class A vessel was chosen because the vessel was carrying high temperature-high-pressure radioactive reactor coolant water. It was a part of the only path for injecting boric acid into the reactor coolant system, and this classification was suggested by then existing Code practice. Evaluation of design codes and system safety classification criteria placed emphasis on safety function and radioactivity release to the environment rather than fluid properties. Therefore, as detailed design progressed, the minimum design requirements for the regenerative heat exchanger were reduced from Class A to Class C. This reclassification of the regenerative heat exchanger conforms to the General Design Criteria requirement that safety related components be designed to quality standards that reflect the importance of their safety function. The unit is located inside the containment building and failure of either the shell or tube side will not result in uncontrolled radioactivity release to the environment, nor prevent safe shutdown of the reactor.

Several additional significant reasons which are considered sufficient justification for acceptance of this classification are as follows:

a. It is possible to isolate the RHX on both the shell and tube side with isolation valves which are remotely operable from the control room and by a manual valve which is outside the containment on the charging side.

b. Should it ever become necessary to completely isolate the RHX, an alternate charging path exists for charging water to the primary plant by utilizing the high pressure safety injection header.

c. Additional Quality Control and Fatigue Analysis requirements have been placed on the RHX, beyond those normally required of an ASME Section III, Class C vessel. These additional requirements are discussed below.

1. A fatigue analysis equivalent to the requirements of a Class A vessel was required of the manufacturer. The analysis has been reviewed under the direction of Licensed Professional Engineers at Combustion Engineering to assure the completeness and accuracy of the analysis.

2. The Quality Control requirements of Appendix IX to ASME Section III have been met. All inspections were performed in accordance with written procedures which had been reviewed by Combustion Engineering (CE) Quality Assurance (QA) personnel. Additionally, CE QA personnel witnessed certain predetermined inspections, and conducted random periodic surveillance inspections. Inspection records have been kept at the office of the manufacturer and also at CE. The certification of inspection compliance has been forwarded to the Omaha Public Power District.

3. Non-destructive testing was witnessed by CE QA personnel who were qualified to ASME Section III, Appendix IX procedures. All nondestructive testing procedures have been reviewed by CE QA personnel and were considered to be in accordance with Appendix IX of ASME Section III.

As a result of the plant design and additional fatigue analysis and QA requirements as discussed above, the present classification of the RHX is considered to be the proper quality design standard for use to satisfy the application of the RHX (Ref. 9.2-2) (Ref. 9.2-3).

9.2.3.2 Letdown Control Valves

The letdown control valves regulate the reactor coolant flow from the regenerative heat exchanger as required by the pressurizer level regulating system. In addition, the valves function to reduce the pressure of the letdown fluid to about 300 psig. The letdown flow is normally about 36 gpm, primarily for coolant purification, but increases or decreases as the pressurizer water level changes. The valves are air operated and fail closed. All parts in contact with reactor coolant are of austenitic stainless steel. The valve characteristics are given in Table 9.2-4 (Ref. 9.2-4).

Table 9.2-4 - "Letdown Control Valves"

Item No's.	LCV-101-1 & 2
Quantity	2
Design Pressure, psig	2485
Design Temperature, °F	650
Flow	
Maximum, gpm	116
Minimum, gpm	26

9.2.3.3 Letdown Heat Exchanger

The letdown heat exchanger cools the letdown stream temperature from the outlet of the regenerative heat exchanger to a temperature suitable for long term operation of the purification system. Component cooling water system fluid is the cooling medium on the shell side of the letdown heat exchanger. Materials of construction are primarily austenitic stainless steel and carbon steel. The characteristics of the letdown heat exchanger are given in Table 9.2-5.

Table 9.2-5 - "Letdown Heat Exchanger"

Item No.	CH-7
Quantity	1
Type	Shell and Tube, Horizontal
Code	ASME III, Class C, 1968
Tube Side (Letdown)	
Fluid	Reactor Coolant
Design Pressure, psig	650
Design Temperature, °F	550
Pressure Loss at 36 gpm, psi	5.5
Pressure Loss at 120 gpm, psi	40
Materials	Austenitic Stainless Steel
Shell Side (Cooling Water)	
Fluid	Component Cooling Water
Design Pressure, psig	150
Design Temperature, °F	250
Materials	Carbon Steel
Maximum Allowable Flow Rate, lb/hr	600,000

Operating Parameters

Tube Side (Letdown)	Normal	Maximum Unbalanced Charging	Maximum Maximum Purification	Unbalanced Letdown
Flow, gpm	36	26	116	116
Inlet Temperature, °F	207	167	375	421
Outlet Temperature, °F	120	120	120	120
Heat Transfer, Btu/hr	1.54×10^6	0.59×10^6	14.6×10^6	17.5×10^6
Shell Side (Cooling Water)				
Flow, gpm	36	11	800	1010
Inlet Temperature, °F	90	50	90	90
Outlet Temperature, °F	177	157	127	125

9.2.3.4 Ion Exchangers

Two mixed bed purification ion exchangers purify the reactor coolant by removing corrosion and fission products. The anion resin is initially in the hydroxyl form and is converted to the borated form. Each unit was designed to handle the maximum letdown flow of 116 gpm. The vessels and retention screens are of austenitic stainless steel construction.

The two deborating ion exchangers are used to remove boron from the reactor coolant when this mode of operation is preferable to a feed and bleed operation. Other than being taller, the units are identical in construction to the purification ion exchangers. The anion resin is initially in the hydroxyl form and is converted to a borated form during boron removal. Each unit is designed for the maximum letdown flow of 116 gpm, and the quantity of resin in each is sufficient to remove the equivalent of 50 ppm of boron from the entire reactor coolant system. The vessels and retention screens are of austenitic stainless steel construction.

The cation ion exchanger is identical in construction to the purification ion exchangers. The unit is charged with a cation resin and is provided for the removal of lithium and cesium fission products from the coolant if required. The unit is designed to handle the maximum letdown flow of 116 gpm.

The characteristics of the ion exchangers are given in Table 9.2-6.

Table 9.2-6 - "Ion Exchangers"

Item No's.	CH-8A, 8B, 9A, 9B & 10 Quantity 5
Type	Flushable
Design Pressure, psig	200
Design Temperature, °F	250
Nominal Operating Pressure, psig	25
Nominal Operating Temperature, °F	120
Resin Volume (usable), ft^3	22
Nominal Flow Rate, gpm	36
Design Maximum Flow Rate, gpm	116
Retention Screen Rating	0.007" x 0.007" opening
Code for Vessel	ASME III, Class C, 1968
Material	Austenitic Stainless Steel
Fluid	6.25 wt % Boric Acid, Maximum

9.2.3.5 Purification Filters

The purification filters collect resin fines and insoluble particulates from the reactor coolant. Each filter can accommodate the maximum letdown flow of 116 gpm. The filter housings are austenitic stainless steel. The characteristics of the filters are given in Table 9.2-7.

Table 9.2-7 - "Purification Filters"

Item No's.	CH-17A & 17B
Quantity	2
Design Pressure, psig	200
Design Temperature, °F	250
Design Flow, gpm	116
Nominal Flow, gpm	36
Maximum Flow, gpm	160
Code for Vessel	ASME III, Class C, 1968
Material	Austenitic Stainless Steel
Fluid	6.25 wt % Boric Acid, Maximum

9.2.3.6 Volume Control Tank

The volume control tank accumulates water from the reactor coolant system. The tank has sufficient capacity within the normal operating level band to allow a full power to zero power decrease without makeup system operation. The tank provides a gas space where a partial pressure of hydrogen is maintained to control the hydrogen concentration in the reactor coolant. A vent to the waste gas vent header permits removal of hydrogen, nitrogen and gaseous fission products released from solution in the volume control tank. The tank is of austenitic stainless steel construction and provided with overpressure protection. Level controls divert coolant to the waste disposal system on high level or can cause a preset blend of concentrated boric acid and demineralized water to be introduced into the tank. However, makeup to the volume control tank is usually controlled manually. The characteristics of the tank are given in Table 9.2-8.

Table 9.2-8 - "Volume Control Tank"

Item No.	CH-14
Quantity	1
Type	Vertical, Cylindrical
Design Pressure, Internal, psig	75
Design Pressure, External, psig	15
Design Temperature, °F	250
Total Internal Volume, ft^3	386
Operating Pressure Range, psig	0 to 65
Nominal Operating Pressure, psig	20 to 35
Nominal Operating Temperature, °F	120
Nominal Spray Flow (letdown), gpm	36
Blanket Gas	Hydrogen and/or Nitrogen
Code	ASME III, Class C, 1968
Fluid	2-1/2 to 4-1/2 wt% Boric Acid, Maximum
Material	Austenitic Stainless Steel

9.2.3.7 Charging Pumps

Three charging pumps are provided to return the purification flow to the reactor coolant system during plant steady state operations. As a result of a Pressurizer Pressure Low Signal (PPLS) or Containment Pressure High Signal (CPHS), all three pumps are started and discharge concentrated boric acid into the reactor coolant system. The pumps are of the positive displacement type. All pressure containing portions of the pump are austenitic stainless steel with internal materials selected for compatibility with boric acid and optimum performance. The charging pumps have a design flow rate of 40 gpm each. The characteristics of the pump are given in Table 9.2-9.

Although the three charging pumps start upon receipt of an Engineered Safeguards signal, no credit is taken for charging pump operation in the USAR Section 14 safety analyses and as such these pumps are not classified as Engineered Safeguards equipment. When the USAR Section 14 safety analyses are more limiting with operation of these pumps, they are assumed to operate.

Table 9.2-9 - "Charging Pumps"

Item No's.	CH-1A, 1B and 1C
Quantity	3
Type	Positive Displacement
Design Pressure, psig	2735
Design Temperature, °F	250
Flow Rate, gpm	40
Nominal Discharge Pressure, psig	2285
Nominal Suction Pressure, psig	50
Nominal Temperature of Pumped Fluid, °F	120
Maximum Discharge Pressure (short term), psig	3010
NPSH Required, psia	7.5
Maximum Pressure Pump Starts Against, psia	2500
Driver Rating, HP	75
Temperature Rise (maximum), °F	15
Materials In Contact with Pumped Fluid	Stainless Steel or Equivalent Corrosion Resistance
Fluid	2-1/2 to 4-1/2 wt % Boric Acid, Maximum

9.2.3.8 Chemical Addition Tank

The chemical addition tank provides a reservoir for the metering pump. The tank is austenitic stainless steel. The characteristics of the tank and the strainer are given in Table 9.2-10.

Table 9.2-10 - "Chemical Addition Tank and Strainer"

<u>Tank, Item No. CH-15</u>

Quantity	1
Capacity, ft^3	1.3
Design Pressure	Atmospheric
Nominal Operating Temperature	Ambient
Material	Austenitic Stainless Steel
Fluid	Hydrazine (N_2H_4); LiOH Hydrogen Peroxide (H_20_2 30% or less)

<u>Strainer, Item No. CH-23</u>

Quantity	1
Type	Basket
Design Pressure, psig	100
Design Temperature, °F	250
Screen Size, US Mesh	60
Design Flow, gph	40
Material	Austenitic Stainless Steel
Fluid	Hydrazine (N_2H_4); LiOH Hydrogen Peroxide (H_20_2 30% or less)

9.2.3.9 Metering Pump

The metering pump is a positive displacement pump constructed of austenitic stainless steel with internals selected for compatibility with various chemicals handled. The pump provides accurately controlled injection of chemicals from the chemical addition tank to the charging pump suction header. The pump characteristics are given in Table 9.2-11.

Table 9.2-11 - "Metering Pump"

Item No.	CH-3
Quantity	1
Type	Positive Displacement, Variable Capacity
Design Pressure, psig	150
Design Temperature, °F	200
Design Flow Rate, gph	40
Design Discharge Pressure (internal relief valve setpoint), psig	100
Nominal Fluid Temperature, °F	Ambient
Horsepower	¼
Materials	Austenitic Stainless Steel
Fluids	Hydrazine (N_2H_4); LiOH; Hydrogen Peroxide (H_20_2 30% or less)

9.2.3.10 Concentrated Boric Acid Storage Tanks

Each of the two concentrated boric acid tanks stores enough concentrated boric acid solution below the normal makeup level band to bring the reactor to a cold shutdown condition at any time during the core lifetime. The solution is prepared in the boric acid batching tank and flows through the boric acid strainer before entering the storage tanks. The combined capacity of the tanks is also sufficient to bring the coolant to refueling concentration before initiation of a cooldown for refueling. The tanks are constructed of austenitic stainless steel. The characteristics of the batching tank, boric acid strainer and the boric acid storage tanks are given in Table 9.2-12.

Table 9.2-12 - "Concentrated Boric Acid Preparation and Storage"

Concentrated Boric Acid Storage Tanks, Item No's. CH-11A & 11B

Quantity	2
Internal Volume, ft^3	773
Design Pressure (internal), psig	15
Design Temperature, °F	200
Type Heaters	Duplicate Electrical Heat Tracing
Fluid, wt % Boric Acid Maximum	2-1/2 to 4-1/2
Material	Stainless Steel
Code	ASME III, Class C, 1968

Boric Acid Strainer, Item No. CH-21

Quantity	1
Type	Basket
Design Pressure, psig	150
Design Temperature, °F	200
Screen Size, US Mesh	60
Design Flow, gpm	50
Materials	Stainless Steel
Fluid	2-1/2 to 4-1/2 wt % Boric Acid

Boric Acid Batching Tank, Item No. CH-12

Quantity	1
Useful Volume, ft^3	67
Design Pressure	Atmospheric
Design Temperature, °F	200
Nominal Operating Temperature, °F	150
Type Heater	Electrical Immersion
Heater Capacity, nomial, kW	36
Fluid	2-1/2 to 4-1/2 wt % Boric Acid
Material	Austenitic Stainless Steel

9.2.3.11 Boric Acid Pumps

The two boric acid pumps supply concentrated boric acid solution to the volume control tank or charging pump suction header, where the boric acid may be diluted with demineralized water. On receipt of the SIAS, these pumps line up with the charging pumps to permit direct introduction of concentrated boric acid into the reactor coolant system. Although the boric acid and charging pumps operate upon receipt of an Engineered Safeguards signal no credit is taken for their operation in the USAR Section 14 safety analyses. Each is capable of supplying boric acid at the maximum demand conditions. Wetted portions of the pumps are austenitic stainless steel with pump internals selected for suitability for the service. The pump and filter characteristics are given in Table 9.2-13.

Table 9.2-13 - "Boric Acid Pumps and Filter"

Pumps, Item No's. CH-4A & 4B

Quantity	2
Type	Centrifugal
Design Pressure, psig	150
Design Temperature, °F	250
Design Head (100 psig), ft	230
Design Flow, gpm	143
NPSH Required, ft	20
Horsepower	30
Fluid	2-1/2 to 4-1/2 wt % Boric Acid
Material in Contact With Liquid	Stainless Steel

Filter, Item No. CH-18 (normally bypassed)

Quantity	1
Type Elements	Synthetic Fiber
Retention of 5 Micron Particles, %	98
Design Pressure, psig	150
Design Temperature, °F	250
Design Flow, gpm	120
Materials	Austenitic Stainless Steel
Liquid	2-1/2 to 4-1/2 wt % Boric Acid
Code	ASME III, Class C, 1968

9.2.4 System Operation

9.2.4.1 Startup

During startup, the plant is brought from cold shutdown to hot standby at normal operating pressure and zero power temperature. During the heatup and after the steam bubble is established, the operator adjusts the pressurizer water level manually with the letdown control valves. The level controls of the volume control tank may automatically divert the letdown flow to the radioactive waste disposal system (RWDS).

If the residual activity in the core is insufficient to reduce the oxygen in the reactor coolant by recombining it with excess hydrogen during heatup, hydrazine may be used to scavenge the oxygen at a coolant temperature below 250°F. If required, chemicals are added to control the pH of the coolant.

The volume control tank is initially vented to the RWDS. After the tank is purged with nitrogen the vent is secured, and a nitrogen and hydrogen atmosphere is established.

Throughout startup, one purification ion exchanger and filter are in service to reduce the activity of the RCS.

While maintaining the required shutdown margin, the boric acid concentration may be reduced during heatup. The operator may inject a predetermined amount of demineralized makeup water by operating the system in the 'Dilute' mode. However, the shutdown CEA groups must be fully withdrawn from the core prior to performing a 'dilution to criticality'. The concentration of boric acid in the reactor coolant is determined by sample analysis.

9.2.4.2 Normal Operation

Normal operation includes operating the reactor both at hot standby and when it is generating power, with the reactor coolant system at normal operating pressure and temperature.

During normal operation:

a. Level instrumentation on the pressurizer automatically controls the volume of water in the reactor coolant system by automatic control of the letdown flow rate and by varying the number of operating charging pumps and pressurizer heaters;

b. Operators control VCT level and boric acid concentration in the RCS by adding concentrated boric acid solution and/or water to the VCT and/or diverting letdown to the RWDS;

c. Instrumentation on the volume control tank may automatically control the level of water in the tank as described in Section 9.2.2. However, this is normally done manually;

d. The hydrogen-nitrogen concentration and pH of the coolant are controlled as described in Section 9.2.2;

e. Changes in reactivity may be compensated for by adjusting the concentration of boric acid in the reactor coolant. Throughout most of the cycle, changes in boron concentration are effected by feed-and-bleed, discharging the excess coolant to the RWDS. Late in cycle life, the dissolved boron in the reactor coolant is maintained at a very low concentration; at this time, the feed-and-bleed method is inefficient, and further reduction is effected by use of the deborating ion exchanger. The makeup system may be operated in four modes:

    1. In the "Dilute" mode, a quantity of demineralized makeup water is selected and introduced into the charging pump suction at a preset rate. When the integrating flowmeter indicates that the selected quantity of makeup water has been added, the flow is automatically terminated;

2. In the "Borate" mode, a quantity of concentrated boric acid is selected and introduced at a preset rate as described above;

3. In the "Manual Blend" mode, the flow rate of the demineralized water and concentrated boric acid are set for any blend concentration between demineralized makeup water and concentrated boric acid;

4. In the "Automatic" mode, the flow rates of the demineralized water and concentrated boric acid are set to achieve the concentration present in the reactor coolant. The solution is automatically blended and introduced into the volume control tank according to signals received from the volume control tank level program. This mode of operation is not normally used.

f. The letdown flow is routed through one of the purification ion exchangers to reduce coolant activity resulting from soluble and insoluble corrosion and fission products. The coolant leaving the purification ion exchanger may be routed through the cation ion exchanger for lithium and cesium removal if necessary.

9.2.4.3 Cooldown

Plant cooldown is accomplished by a series of operations which bring the reactor plant from a hot standby condition at normal operating pressure and zero power temperature, to a cold shutdown.

Before the plant is cooled down, the volume control tank is vented to the RWDS to reduce the activity and the nitrogen-hydrogen concentration in the reactor coolant system. The operator may also increase the letdown flow rate to accelerate degasification, ion exchange and filtration of the reactor coolant. The reactor is shutdown by inserting all the control rods in the reverse order used for startup. Then the operator increases the concentration of boric acid in the reactor coolant to the value required for cold shutdown and verified by chemical analysis prior to commencing plant cooldown. This is done to ensure that the reactor has an adequate shutdown margin throughout its period of cooldown.

During cooldown, the operator uses the letdown control valves and/or the charging pumps to adjust and maintain the level of water in the pressurizer. Makeup water is introduced at the shutdown boric acid concentration. The operator may switch the suction of the charging pumps to the safety injection and refueling water tank (SIRWT). The charging flow may be used as an auxiliary spray to cool the pressurizer when less than three reactor coolant pumps are in operation. This is required because minimal RCP spray flow is available with less than three RCPs in operation.

9.2.4.4 Hot Leg Injection

Long term response to a large break Loss of Coolant Accident (LOCA) requires that, in order to prevent boron precipitation in the core, simultaneous hot and cold leg injection must be initiated. The CVCS system provides a path for hot leg injection in post-LOCA long term cooling. (Ref. 9.2-5).

9.2.5 Design Evaluation

Under emergency conditions, the charging pumps are used to inject concentrated boric acid into the reactor coolant system. Either PPLS, CPHS or pressurizer level control automatically starts all charging pumps. Because this function is not credited in the USAR Section 14 safety analyses these pumps are not considered Engineered Safeguards equipment. The SIAS also transfers the charging pump suction from the volume control tank to the discharge of the boric acid pump. If the boric acid pumps are not operable, boric acid flows by gravity from the concentrated boric acid tank to the charging pump suction header. If the charging line inside the reactor containment building is inoperative, the line may be isolated outside of the reactor containment and concentrated boric acid solution may be injected by the charging pumps through the safety injection system. Containment integrity is maintained during post LOCA situations by maintaining a higher pressure in the charging line than the containment atmospheric pressure.

A CIAS terminates letdown flow by closing two containment isolation valves. During an uncontrolled heat extraction event, the CIAS to the letdown flow isolation valves may be manually overridden (Ref. 9.2-13) for up to one hour in order to reduce excessive RCS inventory. In this situation, it is required that HPSI stop and throttle criteria (as defined in Emergency Operating Procedures) be met prior to overriding CIAS. This use of the letdown system is not credited in the Section 14 safety analyses.

9.2.6 Availability and Reliability

To assure reliability, the design of the CVCS incorporates redundant critical components. This reduces dependence on any single critical component. Redundancy is provided as follows:

Component	Redundancy
Purification Demineralizer	Parallel Standby Unit
Purification Filters	Parallel Standby Unit
Deborating Ion Exchanger	Parallel Standby Unit
Charging Pump	Two Parallel Standby Units
Letdown Flow Control Valve	Parallel Standby Valve
Boric Acid Pump and Tank	Parallel Standby Unit

The charging and boric acid pumps may be powered by the diesel-generators if normal power sources are lost. One charging pump and one boric acid pump are connected to each diesel-generator. The third charging pump may be fed from either emergency diesel (see Section 8). Physical separation and barriers are provided between the power and control circuits for the redundant pumps. Standby features are provided so that at least one charging pump operates after the PPLS and/or CPHS. If both diesels are available, both boric acid pumps operate. The charging pumps and boric acid pumps may be controlled locally at their switchgear. Separate power supplies and control circuits for the pumps assure that the system satisfies the single failure criterion. In the event of a fire in the cable spreading room or control room, one charging pump can be started from the alternate shutdown panel.

When fuel is in the reactor and the reactor is subcritical there shall be at least one flow path to the core for boric acid injection. This flow path may be from the SIRWT, with at least 10,000 gals. available at refueling boron concentration or from a BAST which meets the requirements of Figure 9.2-3 for a SIRWT boron concentration at the technical specification limit.

The minimum volume of borated water contained in the concentrated boric acid tank(s) is dependent on the boric acid storage tank (BAST) and SIRWT boron concentrations. The minimum required volume curve is shown in Figure 9.2-3. Depending on the flow paths available, this volume of borated water can be either the combined volume of the two BASTs, or the minimum in each BAST, or can be contained in a specific BAST. The ambient temperature of the boric acid tank solution CH-11A and CH-11B shall meet the temperature requirements of Figure 9.2-4.

Each concentrated boric acid tank containing 2.5-4.5 weight percent boric acid has sufficient boron to bring the plant to a cold shutdown condition. Boric acid pumps are each of sufficient capacity to feed all three charging pumps at their maximum capacity.

The concentrated boric acid storage tank is sized for 2.5-4.5 weight percent boric acid solution and is capable of storing solution up to 4.5 weight percent solution. All components of the system are capable of maintaining 4.5 weight percent solution. The elevation of the concentrated boric acid tank is sufficiently above the charging pump suction so as to provide adequate gravity flow to the charging pumps. Figure 9.2-4 contains a 10°F bias to account for temperature measurement uncertainty.

The boric acid solution is stored in insulated tanks and is piped in insulated lines to preclude precipitation of the boric acid. If the boric acid pumps are not available, boric acid from the concentrated boric acid tanks may be gravity fed into the charging pump suction. If the charging line inside the reactor containment building is inoperative, the charging pump discharge may be routed via the safety injection system to inject concentrated boric acid into the reactor coolant system.

The piping systems and equipment which are of safety significance are defined on P&ID E-23866-210-120 and E-23866-210-121.

Analyses of pressure pulses in the charging line have been made to provide a basis for the design and installation of accumulators at the charging pump suctions and discharges. These accumulators restrain dynamic loadings on the piping to design limits. (Ref. 9.2-6).

The inputs to the analyses were pressures as a function of time at the pump suction and discharge. The procedures developed by the accumulator manufacturer were then used to obtain the proper size and precharge pressure for the accumulators and the resulting, reduced pressure pulses in the piping system.

9.2.7 Tests and Inspections

All equipment was subject to the test and inspection requirements of the applicable codes. System operation was demonstrated during the preoperational test program.

9.2.8 Specific References

9.2-1 PWR Primary Water Chemistry Guidelines, Electric Power Research Institute, R. A. Shaw, Report Number NP-4762-SR

9.2-2 CE Letter, CE-750-2091, Revision of AEC-Fort Calhoun Station Question 9.9, April 26, 1971, WIP Number 005322.

9.2-3 CE Letter, CE-750-1127, Regenerative Heat Exchanger Code Classification, May 29, 1969, WIP Number 005211.

9.2-4 Engineering Specification for Pneumatic Operated Control Valves, Specification Number 23866-220-704, June 27, 1968, WIP Number 070321.

9.2-5 NRC Safety Evaluation Report Related to the Effect of HPSI Header Cross Connect Valves on Hot Leg Injection, January 10, 1989.

9.2-6 SGA Pulsation Analysis Study, Project 04-4044-035, December 26, 1974, WIP Number 007839.

9.2-7 Specification 23886-220-302, “Engineering Specification for a Shell and Tube Heat Exchanger,” (Regenerative Heat Exchanger, Reference USAR Section 9.2.1.2).

9.2-8 Specification 23886-220-303, “Engineering Specification for a Shell and Tube Heat Exchanger,” (Letdown Heat Exchanger, Reference USAR Section 9.2.1.2).

9.2-9 Specification 750S-2305, “Engineering Specification for Primary Coolant Pipe and Fittings for Omaha Public Power District Fort Calhoun Station,” (Charging Nozzles, Reference USAR Section 9.2.1.2).

9.2-10 ABB/CE Calculation O-PENG-CALC-009, Revision 00 (Regenerative Heat Exchanger, Reference USAR Section 9.2.1.2).

9.2-11 ABB/CE Calculation O-PENG-CALC-010, Revision 00 (Charging Nozzle, Reference USAR Section 9.2.1.2).

9.2-12 ABB/CE Letter O-PENG-99-004, (Letdown Heat Exchanger, Reference USAR Section 9.2.1.2).

9.2-13 NRC Safety Evaluation Report related to Amendment 191 concerning an Unreviewed Safety Question (USQ) on overriding the containment isolation actuation signal closure signal, July 22, 1999.

9.2-14 ABB/CE Nuclear Steam Supply System Chemistry Manual, CENPD-28, Revision 4.

9.2-15 ABB/CE Letter F-PENG-006, L-PENG-004, dated October 3, 1997, Response to Questions Regarding Hydrogen Peroxide Injection at St. Lucie

9.2-16 Florida Power and Light Inter-Office Correspondence ENG-SPSL-98-0567, dated October 16, 1998 (Addition of Hydrogen Peroxide to the RCS during Shutdown Cooling at St. Lucie, Units 1 and 2.

9.2.9 General References

9.2.9.1 Combustion Engineering CVCS System Description, CE-750-1331, September 9, 1969, WIP Number 018040, Revision 0.

9.2.9.2 NRC Safety Evaluation Report Related to Amendment Number 131 to Facility Operating License No. DPR-40, May 18, 1990.

9.2.9.3 NRC Safety Evaluation Report Related to Amendment 172 to Facility Operating License No. DPR-40, December 12, 1995.

9.5 REFUELING SYSTEM

9.5.1 Design Bases

9.5.1.1 General

The refueling system provides for the storage and safe handling of fuel under all foreseeable conditions, from receipt of unirradiated fuel at the plant to shipment of irradiated fuel following radioactive decay. The design and construction of the system includes interlocks, travel and load limiting devices and other protective measures to minimize the possibility of mishandling or equipment malfunction that could damage the fuel and cause fission product release. Power operation of the system components is supplemented by manual backup to insure that the transfer of a fuel assembly can be completed in the event of a power failure.

9.5.1.2 Prevention of Criticality During Transfer and Storage

The transfer canal and pools for handling and storage of the fuel assemblies are filled with borated water in which the boron concentration is maintained at that value which will maintain the core at a k_{eff} of 0.95 or less with all CEAs withdrawn from the core.

Spent fuel assemblies are stored in stainless steel racks consisting of vertical cells grouped in parallel rows with a center-to-center distance in Region 1 of 9.821" (E-W) x 10.363" (N-S) and 8.652" in both directions in Region 2. Both types of racks contain the neutron poison material Boral™. The Boral™ is attached as panels between each storage cell. The panels are protected with a stainless steel sheath. The racks are of rugged design to provide protection aganist mechanical damage to the fuel and the spacing is such that it is impossible to insert assemblies in other than the prescribed locations and to store more than a safe quantity of fuel. Borated water surrounds the spent fuel storage racks at the same concentration as, and to a level common with the refueling cavity and pool. The center-to-center distance of the storage racks is such that a k_{eff} of less than 0.95 is maintained even in the event that the boron concentration is reduced to 500 ppm (Ref 9.5-6).

The spent fuel storage racks consist of two distinct regions. Region 1 can accept either new or irradiated fuel. Region 2, however, can accept only spent fuel meeting the minimum exposure requirements currently specified in Figure 2-10 of the Technical Specifications or if a full length CEA is inserted into the fuel assembly. The criticality analysis with CEA insertion assumed full CEA insertion during their residence in the core. This is conservative since the CEAs had only limited insertion during their exposure in the core. Even with full length insertion in the core it was still shown that the amount of B^{10} remaining in the CEA was adequate to prevent exceeding the NRC criteria. A clip is attached to tie the CEA and fuel assembly together. The clip was designed such that it would not be able to be removed by the grapple on the fuel handling machine under normal handling conditions (Refs 9.5-4, 9.5-6, 9.5-8, 9.5.9.1).

Fuel can be moved directly from the reactor core or from Region 1 to Region 2 of the spent fuel pool after both a review and an independent verification of burnup adequacy have been performed. The fuel burnup determination is performed by surveillance test prior to fuel movement into Region 2.

The new fuel assemblies for a 1/3 core load are stored dry in rigid racks. The new fuel storage rack area was designed to be located on an elevated balcony above the general floor area to preclude flooding.

9.5.1.3 Fuel Storage Radiation Shielding

Adequate shielding for radiation protection of personnel is provided by the handling of irradiated fuel under not less than 10 feet of water. Mechanical stops are provided on all handling equipment which limit the height of withdrawal of the irradiated fuel to maintain the low level of radiation required for unrestricted occupancy of the area by personnel. The system is designed such that water cannot drain by gravity out of the fuel storage pool below the level of the top of the stored fuel in its storage rack (see Section 9.5.3.2).

9.5.1.4 Protection Against Radioactivity Release

Protection against the accidental release of radioactivity from irradiated fuel to the atmosphere is provided by the auxiliary building ventilation system, and by the containment air recirculation, cooling and iodine removal system.
The ventilation air for both the containment and the spent fuel pool area flows through absolute particulate filters and radiation monitors before discharge at the ventilation discharge duct. In addition, the exhaust ventilation ductwork from the spent fuel storage area is equipped with a charcoal filter which will be manually brought on the line whenever spent fuel is being handled. This filter will absorb gaseous iodines in the unlikely event of a fuel handling incident resulting in the release of large quantities of radioactivity. (See Section 9.10.2 and 14.18)

9.5.1.5 Control Room Habitability

Protection from the accidental release of radioactivity for the control room operators is provided by the control room charcoal filters, VA-64A and VA-64B. One of the control room charcoal filter units is started prior to any irradiated fuel movement in the containment building or in the auxiliary building spent fuel pool storage area. New, unirradiated fuel movement does not affect control room habitability, unless it is moved over irradiated fuel, therefore, the control room charcoal filters need only be operated when new fuel is moved over irradiated fuel. Control Room filters need not be operated when new fuel is moved in the receiving/inspection storage area.

9.5.1.6 Spent Fuel Storage Rack Seismic Design

The spent fuel rack is a seismic category I structure. The design of the fuel racks is in compliance with the requirements of USNRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications", Section IV, and Standard Review Plan (SRP) (Ref 9.5-4). The rack is a free-standing structure consisting of discrete storage cells which are loaded with free-standing fuel assemblies. The response of a rack module to seismic inputs is highly nonlinear involving a complex combination of motions (sliding, rocking, twisting, and turning), resulting in impacts and friction effects. Linear methods such as modal analysis and response spectrum techniques cannot accurately simulate the structural response of such a highly nonlinear structure to seismic excitation. A correct simulation is obtained only by direct integration of the nonlinear equations of motion using actual pool slab acceleration time-histories to provide the loading. Therefore, as an initial step in spent fuel rack qualification, four sets of synthetic time-histories for three orthogonal directions are developed in compliance with the guidelines of Rev. 2 of USNRC SRP 3.7.1 and 3.7.2. In particular, the synthetic time-histories must meet the criteria of statistical independence and enveloping of the design response spectra.

To demonstrate structural qualification, it is required to show that stresses are within allowable limits and that displacements remain within the constraints of the contemplated design layout for the pool. This implies that impacts between rack modules, if they occur, must be confined to locations engineered for this purpose, such as the baseplate edge and possibly the upper region of the rack above the active fuel region. Similarly, rack-to-pool wall impacts, are engineered into the rack design and must be within stipulated limits. Impact loads between pedestal and liner must be assessed to assure liner integrity.

Rack dynamic simulations were performed in 3-D single rack 22-DOF model and in Whole Pool Multi-Rack (WPMR) model considering all 11 racks in the pool, respectively. A total of 24 single rack runs were performed for the heaviest rack in the pool, the rack with the maximum ration of side length and a Region I rack considering different values of friction coefficients and fuel loading patterns and assuming the worst earthquake (MHE). Two WPMR runs were performed assuming random distributed friction coefficients subjecting MHE and DE, respectively. All seismic analyses were performed based on both dry weight of 1380 lbs per regular fuel assembly and 2480 lbs per cell for future consolidation considerations.

A three-dimensional ANSYS finite element fatigue analysis was carried out on the representative new spent fuel rack which experiences the highest vertical support pedestal loading as characterized by the Whole Pool Multi-Rack analysis. Stress cycles are characterized by the cyclic life of the direct compressive load and the two friction loads acting on the pedestal. Bounding loads and number of cycles are obtained by examination of the relevant load time-histories for the pedestal chosen for detailed examinations. The number of imposed seismic events is 1 MHE and 10 DE.

The cumulative damage factor calculated to be 0.585 in conformance with the ASME Code indicates that is less than the Code allowable value of 1.0.

9.5.1.6.1 The 3-D 22-DOF Singles Rack Model

a. The fuel rack structure is rigid; motion is captured by modeling the rack as a twelve degree-of-freedom structure. Movement of the rack cross-section at any height is described by six degrees-of-freedom of the rack base and six degrees-of-freedom at the rack top. Rattling fuel assemblies within the rack are modeled by five lumped masses. Each lumped fuel mass has two horizontal displacement degrees-of-freedom. Vertical motion of the fuel assembly mass is assumed equal to rack vertical motion at the baseplate level. The centroid of each fuel assembly mass is located off center, relative to the rack structure centroid at that level, to simulate a partially loaded rack.

b. Seismic motion of a fuel rack is characterized by random rattling of fuel assemblies in their individual storage locations. All fuel assemblies are assumed to move in-phase within a rack. This exaggerates computed dynamic loading on the rack structure and therefore yields conservative results.

c. Fluid coupling between rack and fuel assemblies, and between rack and wall, is simulated by appropriate inertial coupling in the system kinetic energy. Fluid coupling terms for rack-to-rack coupling are based on opposed-phase motion of adjacent modules.

d. Fluid damping and form drag is conservatively neglected.

e. Sloshing is negligible at the top of the rack and is neglected in the analysis of the rack.

f. Potential impacts between rack and fuel assemblies are accounted for by appropriate "compression only" gap elements between masses involved. The possible incidence of rack-to-wall or rack-to-rack impact is simulated by gap elements at top and bottom of the rack in two horizontal directions. Bottom elements are located at the baseplate elevation.

g. Pedestals are modeled by gap elements in the vertical direction and as "rigid links" for transferring horizontal stress. Each pedestal support is linked to the pool liner by two friction springs. Local pedestal spring stiffness accounts for floor elasticity just above the pedestal.

h. Rattling of fuel assemblies inside the storage locations causes the gap between fuel assemblies and cell wall to change from a maximum of twice the nominal gap to a theoretical zero gap. Fluid coupling coefficients are based on the nominal gap.

9.5.1.6.2 Whole Pool Multi-Rack (WPMR) Model

The single rack 3-D (22DOF) model outlined in the preceding subsection is used to evaluate structural integrity, physical stability, and to initially assess kinematic compliance (no rack-to-rack impact in the cellular region) of the rack modules. Prescribing the motion of the racks adjacent to the module being analyzed is an assumption in the single rack simulations. For closely spaced racks, demonstration of kinematic compliance is further confirmed by modeling all modules in one comprehensive simulation using a Whole Pool Multi-Rack (WPMR) model. In WPMR analysis, all racks are modeled, and their correct fluid interaction is included in the model.

The presence of fluid moving in the narrow gaps between racks and between racks and pool walls causes both near and far field fluid coupling effects. A single rack simulation can effectively include only hydrodynamic effects due to contiguous racks when a certain set of assumptions is used for the motion of contiguous racks. In a WPMR analysis, far field fluid coupling effects of all racks are accounted for using the correct model of pool fluid mechanics. The external hydrodynamic mass due to the presence of walls or adjacent racks is computed in a manner consistent with fundamental fluid mechanics principles using conservative nominal fluid gaps in the pool at the beginning of the seismic event. Verification of the computed hydrodynamic effect by comparison with experiments is also provided.

The fluid flow model used to obtain the whole pool hydrodynamic effects reflects actual gaps and rack locations.

The friction coefficient is ascribed to the support pedestal/pool bearing pad interface consistent with Rabinowicz's data. Friction coefficients, developed by a random number generator with Gaussian normal distribution characteristics, are imposed on each pedestal of each rack in the pool. The assigned values are then held constant during the entire simulation in order to obtain reproducible results. Thus, the WPMR analysis can simulate the effect of different coefficients of friction at adjacent rack pedestals.

In Whole Pool Multi-Rack analysis, a reduced degree-of-freedom (RDOF) set is used to model each rack plus contained fuel. The rack structure is modeled by six degrees-of-freedom. A portion of contained fuel assemblies is assumed to rattle at the top of the rack, while the remainder of the contained fuel is assumed as a distributed mass attached to the rack. The rattling portion of the contained fuel is modeled by two horizontal degrees-of-freedom.

Thus, the WPMR model involves all racks in the spent fuel pool with each individual rack modeled as an eight degree-of-freedom structure. The rattling portion of fuel mass, within each rack, is chosen to ensure comparable results from displacements predictions from single rack analysis using a 22-DOF model and predictions from 8-DOF analysis under the same conditions.

9.5.1.7 New Fuel Storage Rack Seismic Design

The new fuel storage rack is a seismic Category I structure. Verification of the seismic I capability of the rack has been performed in Specific Reference 9.5-5. The rack is a free standing braced framed steel structure and is anchored to the supporting floor structure of the Auxiliary Building. New fuel assemblies are supported by a three tiered array of concentric funnels which are held in place by the structural frame. The rack loaded with new fuel, its anchorage, and supporting floor structure were analyzed using the finite element structural analysis program GTSTRUDL. The analysis demonstrated that these systems will not fail nor collapse assuming a worst case earthquake (MHE).

9.5.2 System Description

Refueling is accomplished by handling irradiated fuel assemblies underwater at all times. The refueling cavity and spent fuel pool are filled with borated water to a common level during refueling. The use of borated water provides a transparent radiation shield, a cooling medium, and a neutron absorber to prevent inadvertent criticality.

The refueling system provides a mechanism for transferring the fuel assemblies between the refueling cavity and the spent fuel storage pool through the transfer tube. The storage pool was originally designed to accommodate 178 fuel assemblies (1-1/3 cores) and the spent fuel shipping cask. The capacity of the spent fuel storage pool was subsequently increased to 483 fuel assemblies (3-2/3 cores). In 1983, the spent fuel pool storage capacity was increased to 729 fuel assemblies (5½ cores). In 1994, the spent fuel pool storage capacity was increased to 1083 fuel assemblies (8 cores). Spent CEAs are stored in the spent fuel assemblies.

The refueling machine removes a spent fuel assembly from the core and transports it to the transfer carrier which is in the vertical position. The carrier is then rotated from the vertical position to the horizontal position by the upending mechanism and is moved through the transfer tube to the spent fuel storage pool by the transfer mechanism. The carrier is then rotated to a vertical position and the spent fuel assembly is removed and placed in a storage rack by the spent fuel handling machine. The spent fuel handling machine is designed to remove the fuel from the storage rack and deposit it in a shipping cask for off-site shipment.

During all handling operations, a sufficient water shield is maintained over the top of the fuel assembly to restrict radiation exposures to operating personnel. The refueling water boron concentration is checked periodically to ensure adequate shutdown margins. A minimum concentration of 500 ppm boron must be maintained in the spent fuel pool to prevent criticality during accident conditions (abnormal location of a fuel assembly; Ref. 9.5-6). The pool is normally maintained at the refueling boron concentration. This concentration can vary from cycle to cycle. The current cycle requirements are noted in Section VI of the Technical Data Book.

New fuel assemblies are stored dry in the new fuel storage area. These are provided with vertical racks to hold 48 replacement assemblies. One cell is prohibited from being used as a fuel storage cell, see Table 9.5-1. These racks are designed to preclude criticality even if flooded (Ref. 9.5-1). New fuel assemblies are transported from the storage rack to the new fuel elevator by means of the spent fuel handling machine (FH-12). The new fuel elevator receives the fuel assembly in its raised position and then travels to the bottom of the fuel pool. The fuel assembly is then picked up by the spent fuel handling machine for transportation to the transfer carriage or spent fuel rack. The layout of the refueling system in Containment is shown in Figure 9.5-1. Selected fuel storage data is provided in Table 9.5-1.

9.5.3 System Components

9.5.3.1 Refueling Cavity

The refueling cavity is a reinforced concrete structure with a stainless steel liner that forms a pool above the reactor. During refueling, the cavity is filled with borated water to a depth which limits the radiation at the surface of the water attributed to irradiated fuel in the core to ALARA.

To prevent leakage of refueling water from the cavity, the flange of the reactor vessel is temporarily sealed to the bottom of the refueling cavity. The seal is installed after reactor cooldown but prior to the removal of the reactor vessel head and flooding of the reactor cavity.

The reactor cavity also provides storage space for the upper guide structure, irradiated in-core instrumentation, miscellaneous refueling tools and the core support barrel when its removal is required. The reactor vessel head and the missile shields are stored on the operating floor.

9.5.3.2 Spent Fuel Storage Pool

The spent fuel storage pool is located outside the containment at the west side of the auxiliary building. The pool was designed for the underwater storage of spent fuel assemblies and CEAs after their removal from the core. Decay heat is removed by the cooling system described in Section 9.6. The pool was constructed of reinforced concrete and the entire wetted surface is lined with stainless steel plate. It was designed to support all dead and live loads including hydrostatic loads, temperature gradients, and the effects of tornadoes and the maximum credible earthquake. Drainage grooves are provided behind the stainless steel liner to permit detection of any liner leakage.

Design of the pool and its cooling system and connections to the pool are such that the pool cannot be drained below the level of the top of the stored fuel when in its storage rack. The top of a fuel assembly in a storage rack is about the same elevation as the bottom of the gate connecting the pool with the fuel transfer canal which is at elevation 1008'-6". A plate has been installed across the bottom of the gate opening to raise the minimum possible water level in the pool to 1009'-8.5". There are no pipes in the pool below this elevation. The water inlet line enters the pool at elevation 1034'-0" and terminates at elevation 1031'-7". The drain line enters the pool and terminates at elevation 1011'-4". When AC-187 located just outside the pool wall on the lower suction line is closed, the pool is protected from any ruptures of the line beyond the valve. The line is designed such that a rupture between the valve and the pool wall could not drain the pool below elevation 1011'-0".

Spent fuel assemblies are handled by underwater tools which are operated from a platform on the spent fuel handling machine. The tools are attached to the hoist by a safety connection and are stored in fixtures on the pool wall and the bridge when not in use.

9.5.3.3 New Fuel Storage

The new fuel is stored dry in racks in the auxiliary building conveniently located for receiving and transferring of these assemblies. New CEAs are inserted in the new fuel or dummy fuel assemblies for storage. Calculations confirm that the new fuel storage vault in the dry condition can safety accept fuel enriched up to 5.0% with a K_{eff} well within the current Technical Specification limit, Ref. 9.5-1.

9.5.3.4 Major Handling Equipment

The handling equipment includes the head lift rig, the internals lift rig, the refueling machine, the upending machines, the transfer carriage, the new fuel elevator, the spent fuel handling machine, and the associated controls and communication equipment. Details of the construction, design and function of this equipment are presented in Section 9.5.4.

Table 9.5-1 - "Refueling Equipment Data"

<u>New Fuel Storage Rack</u>	
Core Storage Capacity	1/3
Equivalent No. of Fuel Assemblies	47*
Center-to-Center Spacing of Assemblies, inches	16
<u>Spent Fuel Storage Pool</u>	
Core Storage Capacity	Region 1 - 1.2 Cores Region 2 - 6.9 Cores
Equivalent No. of Fuel Assemblies	1083
Number of Space Accommodations for Spent Fuel Shipping Casks	1
Center-to-Center Spacing for Assemblies, inches	Region 1 - 10.363" x 9.821" Region 2 - 8.652"
Maximum K_{eff} Without Borated Water	Ref. 9.5-6
<u>Miscellaneous Details</u>	
Wall Thickness of Spent Fuel Storage Pool, ft	2 to 5-1/2
Weight of Fuel Assembly, lb (rack design)	1380
Capacity of SIRW Tank, gal	314,000
Quantity of Water Required to Fill Refueling Cavity, gal	249,500
Spent Fuel Storage Pool Volume, gal	215,000

* Rack cell B2 is prohibited from being used as a storage cell.

9.5.4 System Operation

9.5.4.1 Reactor Vessel Head Lifting Rig

The head lifting rig is composed of a removable three-part spreader bar assembly, a three-part column assembly which is attached to the seismic support skirt, and the rigging necessary to lift and move the head to the storage area. The column assembly provides a working platform for personnel during maintenance, supports the three hoists which are provided for handling the hydraulic stud tensioners, the studs, washers and nuts, and links the spreader bar assembly with the seismic support skirt and the head.

9.5.4.2 Internals Lifting Rig

The internals lifting rig consists of an instrument elevator assembly and a structural lifting rig for attachment to the main containment building crane. Three spreader arms are supported by the lift rig in delta arrangement; pipe legs are attached to these spreader arms.

To lift the upper guide structure for refueling, each pipe leg carries a hollow attachment bolt which is threaded into the upper guide structure. In the event that removal of the core support barrel is required, an additional bolt is inserted through each upper guide structure attachment bolt and threaded into the core support barrel; the upper guide structure and the core support barrel are lifted and removed together.

The function of the instrument elevator assembly is to attach the in-core instrumentation retraction structure to the building crane; it also serves as an access platform for instrument line servicing during refueling.

9.5.4.3 Refueling Machine

The refueling machine is a traveling bridge and trolley which spans the refueling cavity and moves on rails located on the 1038'6" elevation floor of the containment building. The bridge and trolley motions allow coordinate location of the fuel handling mast and hoist assembly over the fuel in the core. The hoist assembly contains a coupling device which, when rotated by the actuator mechanism, engages the fuel assembly to be removed. The hoist assembly is moved in a vertical direction by a cable that is attached to the swivel top of the hoist assembly and runs over a sheave on the hoist cable support to the drum of the hoist winch. After the fuel assembly is raised into the hoist and the hoist into the refueling machine mast, the refueling machine transports the fuel assembly to another location or to the carrier assembly.

The controls for the refueling machine are mounted on a console located on the refueling machine trolley. Coordinate location of the bridge and trolley is indicated at the console by digital readout devices which are driven by encoders coupled to the guide rails through rack and pinion gears. A system of pointers and scales is provided as a backup for the remote positioning readout equipment. Manually operated hand wheels are provided for bridge, trolley and winch motors in the event of a power loss.

During withdrawal or insertion of a fuel assembly, the load on the hoist cable is monitored at the control console to ensure that movement is not being restricted. A zoned mechanical interlock is provided which prevents opening of the fuel grapple and protects against inadvertent dropping of the fuel. A piston operated spreader device is provided which spreads adjacent fuel assemblies within the core to provide unrestricted removal and insertion. This spreader is part of the hoist assembly and can be operated before grappling of the fuel assembly when required. The safety features of the refueling machine are:

a. An anti-collision device on the refueling machine mast which stops bridge and trolley motion. This device consists of a hoop and limit switches to protect the mast from hitting the vessel alignment pins, structures within the refueling cavity or the walls of the refueling cavity;

b. Interlocks which restrict simultaneous operation of either the bridge or trolley and the hoist winch drive mechanism;

c. An interlock which prevents bridge and trolley motion when the spreader device is extended;

d. An interlock which prevents bridge and trolley motion when the hoist is loaded and below the hoist box down stop;

e. An override switch which must be actuated after fuel hoist operation to allow bridge or trolley motion;

f. Over and underload switches which stop fuel hoist motion;

g. Underload Bypass switch to allow lowering an empty hoist box;

h. Bridge and trolley speed restriction zones over the reactor core;

i. Fuel hoist speed restriction;

j. An interlock which prevents positioning of the refueling machine over the upending machine unless the upending machine is in the vertical position.

9.5.4.4 Upending Machines

Two upending machines are provided, one in the containment building and the other in the fuel storage area. Each consists of a structural steel support base from which is pivoted an upending straddle frame which engages the two-pocket fuel carrier. When the carriage with its fuel carrier is in position within the upending frame, the pivots for the fuel carrier and the upending frame are coincident. Hydraulic cylinders attached to both the upending frame and the support base rotate the fuel carrier to the vertical position and then to the horizontal position as required by the fuel transfer procedure.

Interlocks are provided to ensure the safe operation of this equipment by preventing inadvertent rotation of the tilting cylinders while FH-1 is in the upender zone, and by deactivating the cable drive so that a premature attempt to move the carriage through the transfer tube cannot be initiated.

9.5.4.5 Transfer Carriage

A transfer carriage is provided to transport the fuel assemblies between the refueling pool and the fuel storage area. Eight large wheels, four on each side support the carriage and allow it to roll on tracks within the transfer tube. Track sections at both ends of the transfer tube are supported from the pool floor and permit the carriage to be properly positioned to the upending mechanism. The carriage is of sufficient length so that it can be driven from one end of the transfer system by means of steel cables connected to the carriage and hence through sheaves to its driving winch mounted at elevation 1056'-8". A two-pocket fuel carrier is mounted on the carriage and is pivoted for tilting by the upending machines.

9.5.4.6 Transfer Tube and Isolation Valve

The fuel transfer tube connects the refueling pool with the fuel storage areas. During reactor operation, the transfer tube is closed by an isolation valve outside the containment and a blind flange inside the containment. The tube is supported by a larger diameter pipe which, in turn, is sealed to the containment envelope. The two concentric tubes are sealed to each other with a bellows expansion joint (see Figure 5.9-9).

9.5.4.7 Transfer Rails

The transfer carriage rides on the transfer rails when moving between the reactor cavity and fuel storage area. The rail supports sit on and are welded to the transfer tube. The rail assemblies were fabricated to a length which allowed them to be lowered for installation in the transfer tube. A gap is left in the track at the valve on the fuel storage side of the transfer tube to allow closing of the valve.

9.5.4.8 Spent Fuel Handling Machine

The basic structure of the handling machine is a traveling bridge which spans the spent fuel pit and moves on rails so as to provide area coverage for both new and spent fuel storage positions, the new fuel elevator and the transfer system tilt machine. A trolley and fuel hoist are mounted on the bridge structure and travel horizontally at 90° to the bridge travel. The hoist hook supports either of two handling tools for grappling fuel assemblies. In operation, the hoist hook and tool are located over the fuel assembly by rectangular coordinate positioning of the bridge and trolley. Grapple load indication and limits are provided to prevent unusual loads during removal and insertion operations. The rotation of fuel is manually controlled via the grapple tool. A portable handling tool is used to install, remove, and reposition Control Element Assemblies (CEAs). Manipulation of the CEAs is performed in the spent fuel pool, with the operator working from spent fuel handling machine FH-12. The same handling tool is also used for manipulation of flow plugs.

9.5.4.9 New Fuel Elevator

A fuel elevator is provided to lower new fuel from the operating level at the top of the transfer canal to the bottom of the canal where it is grappled by the long grapple tool. The elevator is powered by a cable winch and fuel is contained in a simple support structure whose wheels are captured in two rails. New fuel is loaded into the elevator by means of the hoist and short grapple tool.

9.5.4.10 Communications

Direct audible communication between the control room and the refueling machine operator is available whenever changes in core geometry are taking place (Reference 9.5-9). This provision allows the control room operator to inform the refueling machine operator of any impending unsafe conditions detected from the main control board indicators during fuel movement. Direct communication is also provided between the refueling machine operator and the spent fuel pool area.

9.5.4.11 Personnel Safety Features

Safety nets over the fuel transfer canal and the new fuel storage area (Room 25A) provide protection against personnel falling from the 1038' -6" level of the spent fuel deck. The removable safety net over the fuel transfer canal is 5' wide by 21' long. The removable safety net over the new fuel storage area is 25' wide by 21' long.

9.5.5 Design Evaluation

Underwater transfer of spent fuel provides ease and safety in handling operations. Water is an effective, transparent radiation shield and an efficient cooling medium for removal of decay heat. Basic provisions to ensure the safety of refueling operations are:

a. Gamma radiation levels in the containment and fuel storage areas are continuously monitored (see Section 11.2.3 and Reference 9.5-9). These monitors provide an audible alarm at the initiating detector and in the control room, indicating an unsafe condition. Continuous monitoring of reactor neutron flux, with indication in the control room, provides immediate indication and alarm of an abnormal core flux level.

b. Whenever new fuel is added to the reactor core, the source range neutron flux (count rate) is recorded to verify the subcriticality of the core.

c. To prevent fuel assemblies from swinging into the wall during fuel movement in the spent fuel pool, limit switches provide a means of enforcing low speeds around the perimeter of the pool as well as providing bridge and trolley lockouts to prevent collision with the walls of the pool. A Limit Bypass Switch allows travel beyond the lockouts to access the perimeter Fuel Rack locations.

d. The design of the equipment places physical limits on the extent of fuel movement, thereby avoiding the possibility of raising fuel beyond a safe limit. Fuel storage rack spacing provides positive protection against criticality in the event of inadvertent flooding of the fuel storage area with unborated water. The design of the fuel storage pool is such that water cannot drain out of the pool by gravity below the level of the top of stored fuel (see Sec 9.5.3.2) and the elevated new fuel storage area cannot be flooded.

e. In the unlikely event that a spent fuel cask falls into the spent fuel pool, the stainless steel liner could be punctured; however, the 12-foot thick reinforced concrete mat below the pool would not be penetrated and leakage of water from the spent pool would be slow. The spent fuel makeup system can provide 500 gpm and additional water is available from both the demineralized water system and the fire protection system using hoses.

9.5.6 Availability and Reliability

All of the equipment in the system is manually operated, i.e., there are no automatic functions requiring logic control. Manually operated handwheels are provided to allow bridge, trolley and hoist motion in the event of a power loss.

The fuel transfer carriage is longer than the fuel transfer tube, assuring that one end of the carriage is accessible at all times during the transfer operation. Operability of the refueling system is assured by functional testing (to include a load test on fuel handling cranes that will be required to handle spent fuel assemblies), prior to commencing refueling operations.

The applicable fuel handling equipment is tested not more than 14 days prior to moving fuel and is retested thereafter whenever the equipment is idle for more than 14 days. Maintenance which effects interlocks and/or setpoints also requires that the equipment be retested prior to using it to move fuel.

9.5.7 Tests and Inspections

The refueling equipment was partially assembled at the fabricator's facility and each component tested for correctness of operation, after which it was shipped directly to the site where the complete system was installed and tested.

The refueling machine is essentially identical to the Palisades machine which was shipped to Combustion Engineering, Windsor, Connecticut and which satisfactorily completed an extensive acceptance and performance test program.

The refueling machine was mounted on rails over a core and pressure vessel mockup and the upending machines were positioned in an adjacent pit with a simulated transfer tube between them. The following refueling operations were performed:

a. Indexing the refueling machine to the fuel assembly in the core;

b. Engaging and lifting the fuel assembly into the fuel hoist;

c. Indexing the refueling machine to the tilting machine and lowering the fuel assembly into the carriage;

d. Operation of the transfer system, upending the carriage to the horizontal, transferring it through the simulated refueling tube to the spent fuel pool upending machine, and upending the carriage back to the vertical.

9.5.8 Specific References

9.5-1 Criticality Safety Evaluation of the Ft. Calhoun New Fuel Storage Vault, EA-FC-94-029, Rev 0.

9.5-2 Deleted

9.5-3 Deleted

9.5-4 LIC-92-340A Licensing Report for Spent Fuel Storage Capacity Expansion, FLC-92-005, 07/07/92.

9.5-5 EA-FC-95-020, Rev. 0, Seismic and Safety Evaluation of New Fuel Storage Rack.

9.5-6 Criticality Safety Evaluation of the Ft. Calhoun Spent Fuel Storage Rack for Maximum Enrichment Capability, EA-FC-96-01, Rev. 0.

9.5-7 Deleted.

9.5-8 CEA Clips for Locking CEA's to Fuel Assemblies for Region 2 Discharge, EA-FC-92-11, Rev. 0, May 28, 1992.

9.5-9 NRC Amendment 188 to Technical Specifications, December 31, 1998.

9.5.9 General References

9.5.9.1 NRC Safety Evaluation Report Related to Technical Specification Amendment 133, October 2, 1990.

9.5.9.2 NRC Safety Evaluation Report Related to Technical Specification Amendment 13, July 2, 1976.

9.5.9.3 NRC Safety Evaluation Report Related to Technical Specification Amendment 75, September 9, 1983.

9.5.9.4 NRC IE Bulletin No. 78-08, Radiation Levels From Fuel Element Transfer Tubes, June 12, 1978.

9.5.9.5 NRC Safety Evaluation Report Related to Technical Specification Amendment 174, July 30, 1996.

9.5.9.6 NRC Safety Evaluation Report Related to Technical Specification Amendment #155, (Spent Fuel Storage Racks) August 12, 1993.

9.5.9.7 Supplement to NRC Safety Evaluation Report Related to Technical Specification Amendment #155, (Spent Fuel Storage Racks) April 9, 1996.

9.6 SPENT FUEL POOL COOLING SYSTEM

9.6.1 Design Bases

The spent fuel pool cooling system was designed to remove decay heat from spent fuel assemblies stored in the pool and to control and maintain the chemistry and clarity of the pool water. It can remove decay heat from a full core discharged from the reactor 72 hours after shutdown from a power level of 1500 MWt, while maintaining the pool water temperature below 140°F (a heat load of 20.7×10^6 Btu/hr). The pool has the capability to accommodate 1083 unconsolidated fuel assemblies while the reactor is unloaded for maintenance and repairs.

The piping is so arranged that failure of any pipeline connected to the pool will not drain the pool below the top of the spent fuel racks.

The spent fuel pool cooling system was designed and constructed to Class I standards (see Appendix F).

9.6.2 System Description

The spent fuel pool cooling system is shown in P&ID 11405-M-11. The system consists of two storage pool circulation pumps, a storage pool heat exchanger, a demineralizer and filter, two fuel transfer canal drain pumps, piping, valves and instrumentation.

The storage pool pumps circulate borated water through the storage pool heat exchanger and return it to the pool. Cooling water to the heat exchanger is provided by the component cooling water system (see Section 9.7). The purity and clarity is maintained by diverting a portion of the circulated water through the demineralizer and the filter. The fuel transfer canal drain pumps are utilized to:

a. Provide pool make-up water from the safety injection and refueling water (SIRWT);

b. Drain the fuel transfer canal and return the refueling water to the SIRWT or the radioactive waste disposal system (RWDS).

During refueling periods the demineralizer and filter can provide a purification system for the refueling water in the containment refueling cavity. This is accomplished with the reactor coolant drain tank pumps. They take suction from the containment refueling cavity and circulate the borated water through the demineralizer and filter and return it to the spent fuel pool.

While the plant is shutdown, and the core is fully off loaded, the shutdown cooling system provides an emergency backup for the spent fuel pool cooling system in case of failure of that system.

This emergency backup cooling capability of the shutdown cooling system is not available when the CCW or RW systems are out of service for maintenance. This condition is acceptable due to the short duration of the system outages, close attention to the fuel pool heatup rate, and the availability of makeup water sources.

9.6.3 System Components

The design and operating data for the spent fuel pool cooling system components are shown in Table 9.6-1:

Table 9.6-1 - "Spent Fuel Cooling System, Design and Operating Data"

Storage Pool Circulation Pumps, Item No's AC-5A & 5B

Number Installed	2
Type	Horizontal, Centrifugal
Capacity, gpm/pump	900
TDH, ft	120
Nominal Operating Temperature, °F	110
Material of Construction	Austenitic Stainless Steel
Motor Enclosure	Totally Enclosed

Storage Pool Heat Exchanger, Item No. AC-8

Number Installed	1
Type	Shell and U-Tube
Code	ASME Section III, Class C, 1968 & TEMA Class R
Capacity, Btu/hr	9×10^6 to 20.7×10^6*
Materials of Construction	
Shell Side	Carbon Steel
Tube Side	Type 304 SS

Demineralizer, Item No. AC-7

Number Installed	1
Type	Mixed Bed, Non-Regenerative
Code	ASME Section III, Class C, 1968
Flow Rate, gpm	75 to 300
Material of Construction	Austenitic SS

Filter, Item No. AC-6

Number Installed	1
Type	Vertical Cylinder, Non-Back-flushable
Code	ASME Section III, Class C, 1968
Flow Rate, gpm	75 to 300
Retention of 2 Micron Particles, %	95
Material of Construction	Austenitic SS

Fuel Transfer Canal Drain Pumps, Item No's AC-13A & 13B

Number Installed	2
Type	Horizontal, Centrifugal
Capacity, gpm/pump	250
TDH, ft	100
Nominal Operating Temperature, °F	110

*Heat removal capacity varies with pool circulating water temperature conditions. The AC-8 has been rerated as a result of the 1994 spent fuel pool rerack project thermal hydraulic analysis (Reference 9.6.8.1).

Table 9.6-1 (Continued)

Material of Construction	Austenitic SS
Motor Enclosure	Drip-Proof

<u>Spent Fuel Pool Strainer, Item No. AC-14</u>

Number Installed	1
Size, inch	8
Material of Construction	Austenitic SS

<u>Piping</u>

Code	USAS B31.7 1968, Class III and B31.1 1967
Material	ASTM A312, Type 304 SS

9.6.4 System Operation

All system functions are locally controlled. The equipment and instruments are accessible during normal operation. An analysis of fluid samples provides the operator with data for selection and adjustment of flow through the demineralizer-filter circuit. Temperature regulation is accomplished by adjustment of the component cooling water flow to the storage pool heat exchanger. All pumps are started locally.

9.6.5 Design Evaluation

The volume of the spent fuel pool is approximately 215,000 gallons. The system is designed to cool the pool water by recirculating the contents through the cooling loop once every two hours with both pumps operating. The tie to the shutdown cooling system provides a complete and redundant fuel pool cooling loop. The demineralizer-filter circuit can, under normal operating conditions, process one half of the pool contents in 24 hours. To preclude carry-over of resin into the spent fuel pool, the filter is installed downstream of the demineralizer. The demineralizer and the filter can be isolated individually for maintenance purposes, without interruption of the normal cooling operation. To accommodate refueling water purification during refueling periods, the demineralizer and filter have an operating capacity four times as high as the normal requirements. This assures the system's capability to maintain the desired purity and clarity of the pool water.

Make-up to the spent fuel pool, to compensate for evaporation losses, for example, is normally from the SIRWT. Demineralized water can be used for makeup. If necessary, pool water can be pumped to the radioactive waste processing system for disposal.

9.6.6 Availability and Reliability

All the equipment in the system is manually operated; there are no pneumatically or electrically actuated valves. All components are of standard design. Metals in contact with borated water are austenitic stainless steel. The tie to the shutdown cooling system from the spent fuel pool cooling system adds an independent source of fuel pool cooling.*

With a freshly unloaded full core in the pool assumed to be discharged from the reactor 72 hours after shutdown and upon failure of the spent fuel pool cooling system at the peak temperature instant, the pool temperature would rise to the boiling point of 212°F within 7.2 hours. This scenario assumes the spent fuel pool is at the end of usable pool storage life (Cycle 27) with the next cycle's core (Cycle 28) discharged into the pool.

9.6.7 Tests and Inspections

All equipment in the system was cleaned and tested prior to installation in accordance with the applicable codes. The system was also cleaned and hydrostatically tested after installation. Welds were inspected as required by the code and all other connections checked for tightness.

Prior to fuel loading the system was tested with regard to flow paths, flow capacity, heat transfer capability, mechanical operability and purification efficiency. Pressure, temperature, flow and level indicating instruments were calibrated and checked for operability.

The equipment is accessible for inspection and maintenance at all times and is tested periodically in accordance with ASME Section XI Boiler and Pressure Vessel Code.

* This independent source of cooling is not available when the CCW or RW systems are out of service for maintenance. See Section 9.6.2.

9.6.8 References

9.6.8.1 MR-FC-91-009, "Spent Fuel Pool Rerack"

9.6.8.2 NRC Safety Evaluation Report Supporting Amendment Number 13, July 2, 1976

9.6.8.3 NRC Safety Evaluation Report Related to the Modification of the Spent Fuel Pool, September 9, 1983

9.6.8.4 NRC Safety Evaluation Report Related to Ultrasonic Fuel Inspection in the Spent Fuel Pool, March 12, 1987

9.6.8.5 IE Bulletin Number 78-08, Radiation Levels From Fuel Element Transfer Tubes, June 12, 1978

9.6.8.6 Engineering Analysis EA-FC-92-077, Licensing Report for Spent Fuel Pool Storage Expansion

9.7 COMPONENT COOLING WATER SYSTEM

9.7.1 Design Bases

The component cooling water system was designed to cool components carrying radioactive or potentially radioactive fluids. It also serves as a cooling medium for the containment air coolers, steam generator blowdown sampling coolers, and the control room economizer coils. The system provides a monitored intermediate barrier between these fluids and the raw water system which transfers the heat to the river. Thus, the probability of leakage of contaminated fluids into the river is greatly reduced. In the unlikely event of a design basis accident (DBA) the system provides sufficient cooling water to the engineered safeguards systems. System components are rated for the maximum duty requirements that may occur during normal, shutdown or accident modes of operation.

The component cooling water system is an engineered safety features system, and hence, is part of the plant's engineered safeguards, as defined in Section 6. For more supporting details of components the Technical Specifications, the CQE Manual, the EEQ Manual, and the Regulatory Guide 1.97 Responses should be consulted.

The system was designed and constructed to seismic Class I standards (see Appendix F).

9.7.2 System Description

The system is a closed loop consisting of three motor driven circulating pumps, four heat exchangers, a surge tank, valves, piping, instrumentation and controls. The system is continuously monitored for radioactivity which may have leaked into the system flow from the fluids being cooled. Compensatory actions are taken if the radiation monitor is removed from service. The flow diagram is shown in P&ID 11405-M-10 and 11405-M-40.

System volume expansion and contraction due to start-up, shutdown, accidents and load transients are accommodated in the surge tank. The tank is horizontal-cylindrical with a normal water level approximately at the centerline. The upper portion of the tank contains nitrogen overpressure. The pressurization of the component cooling water surge tank exerts a static head on the component cooling water pump suction. In addition, a minimum tank pressure of 34 psig is maintained during normal plant operation to preclude vaporization of CCW in the containment air cooling coils in the event of a loss of offsite power coincident with a LOCA or main steam line break. The pressure can be maintained by two regulating valves, one admitting nitrogen to the tank on falling pressure, the other relieving gas to the vent header through the pressure control valve. Nitrogen is normally isolated to the surge tank and operator action is necessary when pressure needs to be increased. A relief valve, discharging to the RWDS, is provided to protect the system against overpressurization.

The water in the system is demineralized and deaerated and an inhibitor is added for protection against corrosion. Makeup is supplied to the surge tank through a level control valve from the demineralized water system.

Level indicating instrumentation is provided at the tank and in the control room where high and low water level alarms are also annunciated. Heat is transferred from the system to the raw water system in the component cooling heat exchangers. The component cooling water flows through the shell side and the raw water through the tube side. The rejected heat is then discharged by the raw water to the Missouri River.

Equipment to be cooled following a DBA is provided with a redundant cooling water supply directly from the raw water system (see Section 9.8).

During steady state operation, the Component Cooling Water System provides cooling for the following heat loads:

a. Letdown heat exchanger;

b. Reactor coolant pump lube oil and seal coolers;

c. Charging pump oil coolers;

d. CEDM seal coolers;

e. Containment air cooling units;

f. Containment air cooling and filtering units;

g. Sampling heat exchangers;

h. Safety injection tank leakage coolers;

i. Control Room economizer coils;

j. Nuclear detector well coolers;

k. Storage pool heat exchanger;

l. Waste gas compressor seal water heat exchangers;

m. Vacuum deaerator pump heat exchangers.

n. Steam Generator blowdown sample chiller.

Flow paths can be selected from the control room by remote operation of the equipment isolation valves, and by manipulation of local manual isolation valves. Lines penetrating the reactor containment are double valved with open-close position indication in the control room. One valve on each line is located inside and one outside the containment, except for the supply and return lines from the containment air recirculation and cooling system coils, where both isolation valves on each line are installed outside the containment.

9.7.3 System Components

The design and operating data for the component cooling water system equipment are shown in Table 9.7-1.

Table 9.7-1 - "Component Cooling Water System, Design and Operating Data"

Component Cooling Water Pumps, Item No's. AC-3A, 3B & 3C

Number Installed	3
Type	Horizontal, Centrifugal
Capacity, gpm/pump	3425
TDH, ft	210
Operating Temperature, °F	
Nominal (See Section 9.7.4.1)	55 - 110
Post-DBA	$\leq$158
Design Pressure, psig	150
Design Temperature, °F	200
Materials of Construction	
Casing	Cast Iron
Impeller	Bronze or Stainless Steel *
Shaft	Carbon Steel

* - Use of bronze, stainless steel or cast iron is acceptable. Original impellers were cast iron; bronze and stainless steel are acceptable substitutes.

Component Cooling Heat Exchangers, Item No's. AC-1A, 1B, 1C & 1D

Number Installed	4
Type	Shell and Straight Tube
Code	ASME Section III, Class C, 1968 and TEMA Class R
Design Capacity, each, Btu/hr	
Nominal	12.1×10^6
Post-DBA	134×10^6
Design Pressure, psig	150
Design Temperature, °F	300
Materials of Construction	
Shell Side	Carbon Steel
Tube Side	Type 304 SS

Table 9.7-1 (Continued)

Component Cooling Water Surge Tank, Item No. AC-2

Number Installed	1
Type	Horizontal-Cylindrical
Code	ASME Section VIII, 1968
Capacity, gallons	5,200
Design Pressure, psig	50
Design Temperature, °F	200
Material of Construction	Carbon Steel

Corrosion Inhibitor Tank, Item No. AC-15

Number Installed	1
Type	Vertical-Cylindrical
Code	ASME Section VIII, 1968
Capacity, gallons	200
Design Pressure, psig	50
Design Temperature, °F	200
Material of Construction	Carbon Steel

Piping

Code	USAS B31.7, 1968, Class II/III & B31.1, 1967
Material (predominantly)	Seamless, ASTM A-106

9.7.4 System Operation

9.7.4.1 Normal Operation

During normal operation, one-of-three component cooling water (CCW) pumps is in continuous service, while the other two are kept at ready standby. The two standby pumps start automatically in the event the pump in service trips out. The operator then selects one pump to stay on-line, shuts off the other pump, and resets the controls for future automatic start of standby pumps. Pump discharge pressure, flow, and temperature are monitored in the control room. The CCW system is normally operated to minimize perturbations of CCW flow and temperature.

CCW flow is normally maintained through at least two CCW heat exchangers to handle potential CCW pump starts. The number of CCW heat exchangers and raw water (RW) pumps in service during normal plant operation is a function of river temperature and the amount of cooling capability needed to normally maintain CCW temperature between 55°F and 110°F (these temperatures represent the normal operating temperature range of the CCW system, not design limits). For the CCW heat exchangers, "in service" means having both CCW and RW flow. To illustrate the seasonal differences, three-of-four CCW heat exchangers and two-of-four RW pumps are normally in service during the summer when river temperature is high. When river temperature is low during the winter, the normal mode is one-of-four CCW heat exchangers and one-of-four RW pumps in service. When river temperature is below 40°F, a portion of the RW flow can be bypassed through an idle (i.e., no CCW flow) CCW heat exchanger.

Make-up to the component cooling water system is pumped to the surge tank from the demineralized water system through an automatic open-shut valve which is actuated by a level control switch on the surge tank.

Corrosion inhibitors are added to the system at the corrosion inhibitor tank. After the chemicals are placed in the tank and the tank is closed, the component cooling water pumps recirculation line, which normally discharges to the surge tank, is diverted to the inhibitor tank. The chemicals thus transferred to the surge tank and through the surge line are gradually mixed into the component cooling water system.

Flow distribution in the system is monitored in the control room by means of flow and/or temperature indication, and adjustments can be made by remote operation of the valves at various components as required.

A radiation monitor in the pump suction header detects radioactivity which may have leaked into the system (see Section 11.2.3). The system is pressurized with nitrogen and there are no open connections to the building atmosphere. The nitrogen over-pressure control valve discharges to the vent header, and thus any excess gas will be contained in the waste gas system and the pressure relief valve outlets are connected to the waste disposal system, and any outflow, if it occurs, will be processed as a radioactive waste.

9.7.4.2 Shutdown Operation

During shutdown cooling, component cooling water pumps and heat exchangers are placed in service as required to reduce reactor coolant temperature from 300°F to normal refueling temperature, and to maintain the proper reactor coolant temperature during refueling.

In the event a shutdown occurs when the river water temperature is 70°F or above, all cooling loads, except for essential services, may be shut down during the initial cooldown period, when the reactor shutdown load is at its maximum.

9.7.4.3 Post-DBA Operation

Following a safety injection actuation signal (SIAS), the component cooling water system is automatically brought to the following operating conditions if instrument air is available:

a. All three component cooling water pumps are started by the ESF load sequencers (see Section 7.3.2).

b. All four raw water pumps are started by the ESF load sequencers (see Section 9.8.4.).

c. All component cooling water system containment isolation valves except those required to mitigate the accident are closed by a containment isolation actuation signal (CIAS).

d. Component cooling water flow to the spent fuel pool heat exchanger is isolated by Containment Isolation Actuation Signal (CIAS).

e. The valves admitting component cooling water to the safety injection and containment spray pump coolers are opened.

f. The component cooling water and raw water inlet and outlet valves for the component cooling water heat exchangers are opened.

The operator ensures at least two component cooling water pumps and two raw water pumps are operating.

In the event that the instrument air system is not available after a DBA, those CCW system air-operated valves not equipped with safety-related air accumulators may go to their failure positions. Loss of instrument air to these valves may prevent the isolation of CCW flow to non-essential components. In addition, the failure of a single active component is postulated in conjunction with a DBA.

If all normal power sources are lost and only one emergency diesel generator functions, a minimum of one component cooling water pump starts with a second pump available on the swing bus, and two raw water pumps operate, if available (refer to Section 9.8).

Since four component cooling water heat exchangers are normally operable, no functional adjustments are required for their post-DBA operation.

After the contents of the safety injection and refueling water (SIRW) tank are depleted to a preset level, a recirculation actuation signal (RAS) is generated (see Section 7.3.2.7). After the RAS is generated, the valves supplying component cooling water to the shutdown heat exchangers are automatically opened and the safety injection and spray water recirculation mode of operation is established. The plant remains in this mode of operation during the long term post-DBA cooldown period.

The open or closed position of all remote-operated valves, pressures, temperatures, flows and the operation of all pumps associated with the control of a DBA are monitored in the control room.

In the unlikely event of a break or failure of component cooling system equipment leading to a complete loss of cooling water supply, the raw water system can be used to cool engineered safeguard components. However, the failure of an active component and a CCW system break are not both postulated. In addition, a CCW system break is not assumed to occur in the short term after an accident. In the case of a CCW system break, raw water is manually directed via normally handjacked locked closed valves to provide direct cooling of the required DBA controlling equipment (see Section 9.8.2).

Raw water backup cooling capability is not available when the system is out of service for maintenance. This will only occur in mode 5 and shutdown condition 2. This condition is acceptable due to the short duration of the system outages, close attention to the fuel pool heatup rate, and the availability of the makeup water sources.

9.7.5 Design Evaluation

The system is designed as a closed, pressurized, circuit with no venting to the building atmosphere. The total volume of the water in the system is approximately 37,500 gallons. Volume changes due to expansion resulting from changes in the fluid temperature are accommodated in the component cooling water surge tank.

A maximum of four component cooling heat exchangers and any two of the three component cooling water pumps satisfy all the requirements during all modes of plant operation. Post-DBA operations are not limited to one preset mode and further, the safeguard requirements are met with partial systems operation.

The CCW system has sufficient capacity for all normal and shutdown operating modes. In addition, the system is capable of satisfying the design criteria under post-DBA condition with the single failure of an active component and a loss of instrument air. Analyses demonstrate that CCW flowrates to essential equipment would be adequate for removing post accident design basis heat loads. A contribution from the containment air coolers (cooled by CCW) is credited in the mitigation of containment peak pressure for a Main Steam Line Break, but not for a LOCA. Under post-LOCA conditions containment cooling requirements would be met by the operation of the containment spray system. Additional heat removal by the CCW system would increase containment cooldown rates. After a RAS is generated, the CCW flow to the shutdown cooling heat exchangers would be adequate to remove the containment heat loads. The CCW system return temperature would be maintained at or below 158°F under post accident conditions. Heat is removed from the CCW system by the raw water system.

Component cooling water flow is normally maintained through four component cooling heat exchangers when river temperature is greater than or equal to 83°F. This eliminates the potential failure of a CCW valve to open as a credible single active failure.

The minimum required hydraulic performance for a component cooling water pump is calculated based on the credited containment air cooler heat removal rate in the MSLB containment pressure analysis (Ref. 14.16-7).

9.7.6 Availability and Reliability

A maximum of any two of the three component cooling water pumps and four component cooling heat exchangers satisfy all of the systems design criteria. The emergency diesel-generators ensure power supply if the off-site power supply is interrupted and either generator operates sufficient equipment to provide the design post-DBA cooling.

All essential operations are performed from the control room, and during normal operation the equipment is accessible for inspection and maintenance. All containment penetrating lines are double valved with fail safe features. Vital equipment is provided with a redundant cooling water supply from the raw water system. *

* This redundant source of cooling water is not available when the system is out of service. See Section 9.7.4.3.

9.7.7 Tests and Inspections

All the equipment in the system was cleaned and tested prior to installation in accordance with the applicable codes. The system was also cleaned and hydrostatically tested after installation. Welds were inspected as required by the code and all other connections checked for tightness.

Prior to start-up the system was tested with regard to flow paths, flow capacity, heat transfer capability and mechanical operability. The pumps and valves were tested for actuation at the design set points. Pressure, temperature, flow and level indicating and controlling instruments were calibrated and checked for operability.

The equipment, except for the piping inside of the containment, is accessible for inspection and maintenance at all times.

9.8 RAW WATER SYSTEM

9.8.1 Design Bases

The raw water system was designed to provide a cooling medium for the component cooling water system. The system is rated for the maximum duty requirements that may occur during shutdown or accident modes of operation. The heat transferred to the raw water is discharged to the river. The water temperature can vary between 33°F in winter to a maximum of 90°F in summer. For protection against a complete failure of the component cooling system, raw water can be diverted to cool engineered safeguards equipment.

The system was designed and constructed to seismic Class I standards (see Appendix F).

9.8.2 System Description

Four raw water pumps are installed in the intake structure pump house to provide screened river water to the component cooling heat exchangers. The pump discharge piping is arranged as two headers which are interconnected and valved at the pumps and in the auxiliary building. Each header was designed to accommodate sufficient flow to the component cooling heat exchangers to support normal modes of plant operation. System pressures, flows and valve positions are displayed in the control room. Water level instrumentation in the intake structure will alarm in the control room if water from any source should endanger the raw water pumps. A majority of the system operational and control functions can be performed from the control room. The flow diagram is shown in P&ID 11405-M-100.

In the unlikely event of a complete failure of the component cooling water system, raw water direct cooling capability exists for the shutdown cooling heat exchangers, control room air conditioning waterside economizers, safety injection and containment spray pump coolers, and the containment air cooling coils. Raw water direct cooling is utilized via normally handjacked locked closed valves. In the event of a complete loss of component cooling water, raw water direct cooling of the shutdown cooling heat exchangers would be needed for long-term decay heat removal after a large LOCA. Cooling water is not required as a heat sink for the control room air conditioners because they have air-cooled refrigerant condensers. Raw water direct cooling may be used for the control room A/C waterside economizers, if desired. Raw water may be utilized for direct cooling of the safety injection and containment spray pumps if locally accessible. They may be inaccessible post-RAS, however the pumps can perform their post-accident function without cooling water. Raw water direct cooling to the containment air coolers is not required for containment peak pressure suppression. Raw water may be used for direct cooling of the containment air cooling coils if containment atmospheric temperature is less than 150°F. Raw water is credited as backup to CCW for fire events as described in Appendix R, Safe Shutdown Analysis. Raw water is also credited in the Fire Safe Shutdown Analysis and the Seismic Safe Shutdown Analysis equipment list for makeup to the Emergency Feedwater Storage Tank, FW-19.

Raw water direct cooling will not be available when the system is out of service for maintenance. This will only occur in Mode 5 and Shutdown Condition 2. This condition is acceptable due to the short outage duration, close attention to the fuel pool heatup rate, and the availability of makeup water sources.

The two raw water lines between the intake structure and the auxiliary building are buried in separate trenches. At the point where raw water piping enters a building, the detail shown in Figure 9.8-2 is employed. The outside guard pipe absorbs forces imposed by the soil in the event of an earthquake. During an earthquake, differential movement between the auxiliary building and the surrounding earth occurs, since the building is connected to bedrock via piles. The process pipe can flex inside the guard pipe, where it is free of soil reactions, enough to absorb the movement between the building and the surrounding earth.

9.8.3 System Components

The design and operating design characteristics of the major raw water system components are shown in Table 9.8-1.

Table 9.8-1 - "Raw Water System, Design and Operating Data"

<u>Raw Water Pumps, Item No.'s AC-10A, 10B, 10C & 10D</u>

Number Installed	4	
Type	Vertical, Mixed Flow	
<u>Pump characteristics:</u>	<u>Flow, gpm per pump</u>	<u>TDH, ft</u>
Nominal Flow and TDH	5325	118
Design Pressure, psig	150	
Design Temperature, °F	150	
Materials of Construction:		
Bowl	AISI 4330 or approved equivalent	
Impeller	AISI 4330 heat treated or approved equivalent	
Shaft	Type 410 stainless steel	
<u>Piping</u>		
Design pressure	150 psig	
Design temperature	500°F	
Code	USAS B31.7 1968, Class II/Class III and B31.1 1967	
Material	Seamless ASTM A106	

9.8.4 System Operation

9.8.4.1 Normal Operation

The system is remotely operable from the control room. Raw water flow is normally maintained through at least two component cooling heat exchangers. The number of component cooling heat exchangers in service (i.e., having both raw water and component cooling water flow through them) during normal plant operation is a function of river temperature and the amount of cooling capability needed. This is described in more detail in Section 9.7.4.1.

9.8.4.2 Shutdown Operation

During shutdown cooling, raw water pumps and heat exchangers are placed in service as required to reduce the reactor coolant temperature. Refer to Section 9.7.4.2.

9.8.4.3 Post-DBA Operation

In the unlikely event of a DBA all four RW pumps are started automatically and the eight heat exchanger isolation valves are opened from a safety injection actuation signal (SIAS) and the eight heat exchanger isolation valves are opened. Additionally, SIAS override to the RW isolation valves is provided so the operator may close/isolate one or more RW heat exchangers.

The failure of a single active component is postulated in conjunction with a DBA. The most limiting single failure is that which results in the least amount of heat removal capability from the CCW System. If all normal power sources are lost and only one emergency diesel-generator functions, a minimum of two raw water pumps would operate if the river water temperature is greater than 60°F. When the river water temperature is below 60°F, one raw water pump may be in an inoperable condition in accordance with Technical Specification 2.4.1.c. Therefore, a minimum of one raw water pump would operate after a diesel generator failure if the river water temperature is below 60°F.

As discussed in Section 9.7.4.3, the raw water system is manually directed to provide direct cooling of the required DBA controlling equipment if the CCW system is not available due to a break or failure of CCW system equipment. However, the failure of an active component and a CCW system break are not both postulated. In addition, a CCW system break is not assumed to occur in the short term after an accident.

9.8.5 Design Evaluation

The raw water system was designed to provide sufficient flow and head capability to maintain the component cooling water at a maximum return temperature of 110°F during normal operation. In addition, the system is capable of satisfying the design criteria under post-DBA conditions with the single failure of an active component and a loss of instrument air. Analyses demonstrate that raw water flow to the CCW heat exchangers would be adequate for removing post accident design basis heat loads while maintaining a maximum CCW return temperature of ≤158°F.

River level/temperature limits are observed for some off-normal raw water system operating alignments. The river condition limits are based on calculations which use the low-limit hydraulic performance for the raw water pumps. The river condition limits, in conjunction with the minimum required raw water pump performance, ensure that if a DBA occurs while in an off-normal alignment, subcooled conditions will exist in the RW discharge header and CCW return temperature will be at or below the maximum post-DBA temperature specified in Section 9.7. The raw water pump minimum hydraulic performance limit is chosen by OPPD. The river condition limits are a function of the chosen raw water pump performance limit, because the cooling performance of the Raw Water system is a function of river conditions as well as pump hydraulic performance. The river condition limitations are documented in the Technical Data Book.

Raw Water flow is normally maintained through four component cooling heat exchangers when river temperature is greater than or equal to 70°F. This eliminates the potential failure of a RW valve to open as a credible single active failure.

The raw water system is capable of providing direct cooling of the required DBA controlling equipment if the CCW system is not available. Analyses demonstrate that adequate flow would be provided to the required equipment and that the temperature of the raw water returning to the river would be less than 210°F with the river at its peak temperature of 90°F.

9.8.6 Availability and Reliability

The system piping between the intake structure and the auxiliary building was designed as a two 20 inch header system. The discharge piping from the four pumps is manifolded, valved and instrumented to permit operation or isolation of any pump. Sufficient flow is available, under any normal mode of operation, even if one of the two supply headers should rupture. Redundant pumping capacity is provided.

To ensure system integrity in the event of an earthquake, each 20 inch header is encased in a 28 inch cast-in-concrete guard pipe in the south wall of the auxiliary building.

The emergency diesel-generators ensure a power supply if the off-site power is interrupted and either generator operates sufficient equipment to provide the design post-DBA cooling.

Air accumulators inside the intake structure provide instrument air to operate the raw water system valves in the intake structure even upon failure of the compressed air system.

Water level instrumentation in the intake structure will alarm in the control room if water from any source should endanger the raw water pumps.

Protection for the raw water pumps and their drives against floods is provided at three elevations as indicated on Figure 9.8-1. The pumps are permanently protected against any water level up to elevation 1007.5 feet by the Class I concrete substructure of the intake building. Protection is provided to elevation 1009.5 feet by gasketed steel closures which are provided for all openings in the reinforced concrete perimeter wall extending upward from the main operating level. Supplementing the wall with sandbags provides protection to elevation 1014.5 feet. For water levels above 1007.5 feet, the water level inside the intake structure is controlled by positioning the exterior sluice gates to restrict the inflow into the wet walls to match the rate of pumped outflow. Because of the wide head variations possible, the sluice gate and pump settings are automatic self-balancing within reasonable limits.

The following water levels were considered in computing the theoretical river levels:

983.0 feet	During the winter, water releases are normally controlled to maintain this level.
992.0 feet	During the navigation season, this level is required.
998.0 feet	Stage duration level exceeded only 1% of the time.
1001.3 feet	Peak level of a 1% probability flood.
1004.2 feet	Peak level of the designed flood.

Theoretical flood levels:

1009.3 feet	Computed peak level flood resulting from the simultaneous occurrence of:

1. The maximum probable rain storm and runoff downstream from Gavins Point;

2. The maximum outflow from Gavins Point resulting from a maximum probable rain storm and runoff upstream from Gavins Point.

There is a two day lag between the release of a peak flow at Gavins Point and its arrival at the Fort Calhoun Station.

1013 feet to 1014 feet	Approximately computed peak level resulting from the simultaneous occurrence of:

1. The conditions No. 1 and 2 as shown in the 1009.3 peak;

2. A catastrophic, instantaneous disintegration of Fort Randall Dam superimposing all of the water impounded behind it on the conditions described above.

The combination of events resulting in a flood level of elevation 1013-1014 feet is considered incredible. If a catastrophic instantaneous disintegration of Fort Randall Dam should occur, however, there is a three-day time lag before flood arrival at the Fort Calhoun Station.

The intake structure is a massive concrete building set just back of the harbor line of the river. The noses of all of the intake or recirculation channels are armored with anchored steel plates. The river bottom slopes downward from the bank to the thread of the channel, thus keeping boats and barges away from the actual harbor line. Any blow that could be struck by such a vessel would be a glancing one at worst on the armored wall noses and any damage to the structure itself is considered unlikely. Even a flood or storm driven barge which might strike the intake structure could not conceivably block flow sufficiently in the three sections of the structure to decrease the flow from the raw water pumps. The intake structure consists of three bays separated by concrete walls perpendicular to the river; two cells contain one raw water pump each and one cell contains two. Flow through each cell is independent of the other two, and the raw water pumps are located 35 feet back from the river. Icing conditions at the river water entrance of the intake structure are prevented by routing a portion of the warm water from the circulating water discharge tunnel back upstream of the intake screen. Experience with the District's stations on the Missouri River shows that, by controlling the amount of water being recirculated, potential icing conditions can be averted.

The Corps of Engineers adjusts winter releases from Gavins Point as necessary to accommodate the needs of all Missouri River water users. Normally, the water level is maintained higher than 983.0 feet. Although agreement between OPPD and the Corps of Engineers to maintain minimum river water levels has not been formalized, the Corps of Engineers does cooperate with OPPD in these matters and would provide additional flow from upstream dams if such conditions would be impending. The time required for severe ice blockage of the river to occur extends into many hours and the weather conditions which would cause blockage would be evident over a period of a few days. Even lower river water levels would not be detrimental to plant safety since the minimum submergence on the raw water pumps is 976 feet 9 inches or more than six feet below the controlled minimum river water level.

An evaluation has been performed of the flooding consequences of a postulated pipe failure involving a system in or above the raw water pump rooms (fire protection, raw water, screen wash). Since these are not high-energy systems (pressure <275 psig and temperature <200°F), the "postulated pipe failure" is a through-wall pipe crack, with the crack size being half the pipe diameter in length and half the pipe thickness in width. The evaluation shows that for any single postulated pipe failure, ample time is available to implement operator actions to isolate the leak before the raw water pump room water level reaches the raw water pump motors.

In the unlikely event all raw water pumps are unavailable, cooling water can be obtained from the fire protection system and its diesel or electric driven fire pumps to provide some base load cooling until raw water can be restored. This system interconnection would be made between the raw water/component cooling water heat exchangers and local fire hose cabinets. Operator actions required for this interconnection are included in one of the plant's abnormal operating procedures.

Should the raw water system become inoperable, the reactor can be stabilized by removing decay heat through the steam generators.

9.8.7 Tests and Inspections

All the equipment in the system was cleaned and tested prior to installation in accordance with the applicable codes. The system was cleaned and hydrostatically tested after installation. Welds were inspected as required by the code and all other connections checked for tightness.

Prior to startup, the system was tested with regard to flow paths, flow capacity, heat transfer capability and mechanical operability. The pumps and valves were tested for actuation at the design setpoints. Pressure, temperature and flow indicating and controlling instruments were calibrated and checked for operability.

The equipment is accessible for inspection and maintenance at all times.

9.8.8 General References

9.8.8.1 NRC Safety Evaluation Report Related to Amendment Number 120 to Facility Operating License No. DPR-40, December 31, 1988.

9.10 HEATING, VENTILATING AND AIR CONDITIONING SYSTEMS

9.10.1 Design Bases

The heating, ventilating and air-conditioning systems are designed to maintain a suitable environment for equipment and personnel and to protect personnel and the public from airborne radioactivity. In the discussions which follow, uncontrolled access areas are defined as those to which operating personnel have unlimited access during normal plant operation and controlled access areas are defined as those areas subject to potential release of radioactivity. The controlled access area systems (serving the containment and part of the auxiliary building) are designed to handle airborne contaminants so that offsite concentrations and in-plant doses (which are controlled by administrative procedures) are within 10 CFR Part 20 limits. The control room system is also designed to restrict the intake of airborne activity in the event of the design basis accident (DBA).

Total leak rates of 1200, 600 and 300 cc/hr from pump seals, flanges and valves respectively to the controlled access area of the auxiliary building are assumed; these leak rates constitute the basis for the rate of air change in the various areas and compartments in order that 10 CFR Part 20 requirements may be met.

The systems are designed on the basis of outside ambient air temperatures of -11°F in winter and 95°F in summer. A summer design outside air wet bulb temperature of 78°F is used for the control room air conditioning system. The design space temperatures are as follows:

Table 9.10-1 - "Design Space Temperatures"

	Winter (°F)	Summer (°F)
Auxiliary Building		
Controlled Access Area	70	105
Uncontrolled Access Area	70	105
Turbine Building	65	105
Engine Driven Auxiliary Feedwater Pump Room	40	122
Containment		
Main Area (plant operating)	120	120
Main Area (cold plant and purge)	55	100
CEDM Enclosure (seismic skirt)	155	155
Nuclear Detector Wells	110	110
Control Room (@ 50% RH)	78	78
Computer Room (@ 50% RH)	75	75
Radioactive Waste Processing Building		
General Area	70	105
Office Area	72	78
Chemistry and Radiation Protection Building	75	75
Office/Cafeteria Addition	75	75

The temperatures for the controlled access area of the auxiliary building are average figures since temperature is not the sole criterion governing air change rates.

The auxiliary building, containment and control room systems are designed and constructed to Class I standards (see Appendix F).

The criteria for design of the control room air-conditioning system are that system failure will not prevent safe plant shutdown. The main control room air-conditioning system consists of two refrigeration and air handling units. Either unit can be selected for automatic operation if the running unit should fail. During periods of above normal heat load both units can be placed into operation. A VIAS will start both units. However, the third stage cooling is locked out, to minimize diesel loading, until the VIAS signal is overridden at the control room panels AI-106A and AI-106B. The two units design and connection to emergency power sources assures that safety related components in the main control room will not be adversely affected by the loss of either refrigeration or air handling unit. The air cooled condensers located on the auxiliary building roof for the refrigeration units are protected from 360 mph tornado winds. Standard Review Plan (SRP) Section 2.2.3 was used to design the air cooled condensers windscreen. The SRP criteria was met, therefore, no tornado missile shielding for the air cooled condensers is required. Reference Calculation FC06375.

The electronic equipment used in the plant safety related components can operate at 120°F continuously. The portions of the reactor protective system located in the control room were designed to operate up to 135°F and 90% relative humidity.

The heat load within the reactor protective system cabinets is low and the cabinets do not require specific air-conditioning ducting.

Temperature control of safety related instrumentation and controls outside the control room but inside the auxiliary building is maintained by the auxiliary building ventilation system. It is designed to maintain the interior of the auxiliary building below at or 105°F. If air temperature should approach 120°F in any of the rooms in the auxiliary building containing safety related instrumentation and controls, temporary supplementary cooling will be initiated. Air conditioning has been added to the switchgear rooms so that the design maximum temperatures will not be exceeded due to the separation of the redundant switchgear trains by a fire barrier. The two separate HVAC systems for the switchgear rooms are cross-tied to allow for service to either side of the fire barrier if either unit is out of service. Additional air conditioning equipment has been installed in the Counting Room area, due to the addition of temperature sensitive equipment and in the Electrical Penetration Room, due to the high heat loadings.

Temperature inside the containment is maintained below 120°F by the containment ventilating system. The maximum initial temperature assumed in the containment pressure safety analysis (Section 14.16) is 120°F.

The hydrogen purge equipment is part of the containment ventilating system, shown schematically on Figure 9.10-1. The criteria used to design this sub-system are:

a. A daily average purge rate of 25 CFM and a maximum purge rate of 250 CFM shall be possible;

b. Redundant blowers shall be supplied;

c. Control of flow rate shall be by positive means;

d. Design of the system shall recognize variation of pressure differential between the containment interior and exterior due to barometric pressure, gas accumulation, and temperature variations inside the containment;

e. Leakage from the system upstream of the filters shall be minimized;

f. The system shall be tested by periodic operation;

g. Radioactivity discharged through the system shall be measured by the stack monitoring system.

The environmental conditions specified for this system are:

Ambient temperature - 40 to 105°F
Ambient pressure - atmospheric
Radiation - 10 R/hr averaged over 40 year life
Relative humidity - 20 to 100%

The control room charcoal filter system was designed using standard review plan (SRP) 6.4 as guidance. Specifically, the following criteria was used:

a. Isolation dampers - dampers used to isolate the control room zone from adjacent zones or the outside should be leak tight. This includes the inlet and outlet dampers for each of the filter units. All isolation dampers for the control room filter system are zero leakage butterfly valves.

b. Single Failure - a single failure of an active component will not result in loss of the system's functional performance. This is accomplished by providing redundant isolation dampers, filters units, and fans. The recirculation damper is not redundant, however, the repair option delineated in Appendix A of the SRP is exercised and is reflected in the operator dose calculations. (See Section 14.15). Controls for each filter train and associated dampers are located is separate control panels. Power for each of the filter trains and associated controls is provided from separate safety related busses. The loss of a diesel generator will not result in a loss of both filter trains.

c. The control room ventilation system has a pressurization rate greater than or equal to 0.5 volume changes per hour and is subject to periodic verifications (each refueling cycle) that the makeup is ±10% of design value.

d. Credit for iodine removal for the atmosphere filtration system is determined in accordance with the guidelines of Regulatory Guide 1.52. The control room charcoal filter units meet all of the applicable RG-1.52 requirements for a 99% (elemental & organic-iodine removal efficiency.

e. Control room inlets are located considering the potential release points of radioactive material and toxic gases.

f. In accordance with GDC 19, doses to an individual in the control room do not exceed the following for any postulated design basis accident:

   whole body (gamma) : 5 rem
   thyroid (iodine) : 30 rem
   skin (beta) : 30 rem

g. There are no chronic effects from exposure to toxic gases. Acute effects, if any, are reversible within a short period of time (several minutes) without benefit of any measures other than the use of self contained breathing apparatus.

In addition to the above, the following codes/standards apply to the control room charcoal filter system:

Overall system design is in accordance with ANSI N509 - 1980. Initial acceptance and periodic surveillance testing is in accordance with ANSI N510 - 1980. Design and testing of fire protection equipment and piping for the carbon adsorber sections of the control room charcoal filter units is in accordance with NFPA 15 and NFPA 13. Fire detection equipment is installed in accordance with NFPA 72D and 72E. Instrumentation for each filter unit is provided in accordance with SRP 6.5.1.

9.10.2 System Description

9.10.2.1 Auxiliary Building Ventilation System

The auxiliary building is ventilated and cooled with ambient outside air. It is divided into two zones for ventilation purposes, the controlled access area and the uncontrolled access area. Both systems are of the once-through, non-recirculating type using supply and exhaust fans.

Portions of the auxiliary building ventilation system are utilized by the hydrogen purge system, which is an engineered safety features system and hence, is part of the plant's engineered safeguards. The part in question is from penetrations M-30 and M-69 to the stack. For more supporting detail of components the Technical Specifications, the CQE Manual, the EEQ Manual, and Regulatory Guide 1.97 Responses should be consulted.

Controlled Access Area System

The controlled access area ventilation supply system consists of an air handling unit, containing roughing filters and preheat and reheat steam coil banks, two 50 percent capacity vane axial fans and distribution ductwork. The exhaust system consists of three 33-1/3 percent capacity vane axial fans drawing air through return ducts from each ventilated space to a common filtering unit containing high efficiency particulate air (HEPA) filters. The exhaust air is continuously monitored for radioactive contamination at the ventilation discharge duct before discharge to atmosphere. The total air throughput is 72,500 CFM and its distribution is shown on P&ID 11405-M-2.

Air is supplied to the ventilated spaces through multi-blade dampers and exhausted through ducts equipped with butterfly dampers. These dampers are pneumatically operated with remote-manual control. Each separately ventilated compartment can be isolated.

The system was designed and balanced so that the zones of highest potential radioactive contamination are at a negative pressure, relative to adjacent areas, in order to prevent outflow of air.

Charcoal filters are installed in normally bypassed ducts at the exhaust of the safety injection and spray pump rooms and the spent regenerant tank room. These filters could be remotely-manually brought onto line in the event of an accidental release of activity in these rooms during a plant emergency (Ref. 9.10-1). A Ventilation Isolation Actuation Signal (VIAS) opens the supply and return dampers for these three rooms. A differential pressure gauge is installed across each filter to provide a means of determining the condition of each filter.

A charcoal filter is also installed in a normally bypassed section of the return ductwork drawing air from the spent fuel storage pool area. Prior to spent fuel handling, the filter will be placed in service to adsorb gaseous iodines in the unlikely event of a fuel handling incident resulting in the release of large quantities of radioactivity (Section 14.18). A differential pressure gauge is installed across each filter to provide a means of determining the condition of each filter (References 9.10-1 and 9.10-9).

Additionally, temporary carbon filtration may be added to the Auxiliary Building Controlled Access ventilation system exhaust housing as required to reduce iodine releases to the environment.

The principal controls and supervisory instruments are located in the control room (see Section 7.6). The temperature instrumentation and controls are located at the air handling unit. The instrumentation is used to monitor the system exhaust temperature. The controls modulate the heating coil steam valves. In the event of a fault resulting in a low preheat coil exit temperature the controls protect the coils from condensate freezing by tripping the fans.

Uncontrolled Access Area System

The uncontrolled access area system is similar to that in the controlled access area, except that shut-off dampers are not installed, the exhaust is not filtered, and a single roof mounted centrifugal exhaust fan is employed. The total air throughput is 22,500 CFM; its distribution is shown on P&ID 11405-M-2.

The system controls and supervisory instruments are located in the control room. The temperature control system is similar to that of the controlled access area ventilation system.

9.10.2.2 Turbine Building Ventilation System

The turbine building ventilation system consists of four air handling units equipped with outside air intake louvers, recirculation air return dampers, steam heating coils with bypass provision, roll filters and four centrifugal fans which supply air to the building through ductwork. Air is exhausted from the building by fourteen roof mounted fans. The system provides once-through ventilation during the cooling season and operates with partial air recirculation during the heating season. The total fan capacity is 500,000 CFM. Additional local heating is provided by steam unit heaters.

The air handling unit and controls are located in the auxiliary building mechanical equipment room directly over the diesel-generator rooms. This room does not have direct access to the remainder of the auxiliary building, however, a door connecting rooms 81 and 82 does exist.

A ventilation system separate from the turbine building ventilation system is provided to maintain the engine driven auxiliary feedwater pump room at mild environment conditions during pump operation. The system consists of intake and exhaust grills, fire dampers, propeller fans and back draft dampers. Air from the turbine building air space is drawn into the room by two wall mounted fans and by the diesel engine as combustion air. Air is exhausted from the room at a location which minimizes recirculation. No local heating is provided in the room: components are design to withstand maximum and minimum room temperatures.

9.10.2.3 Containment Air Cooling and Ventilation Systems

The containment is served by three separate systems, the containment air recirculation, cooling and iodine removal system, the nuclear detector well cooling system and the containment purge system.

Containment Air Recirculation, Cooling and Iodine Removal System

The system removes heat released to the containment atmosphere during normal plant operation; it is also an engineered safeguards system, and as such is fully described and discussed in Section 6.4. During normal operation filtered air is distributed to the various areas of the containment through ductwork. In an emergency the ductwork is not relied upon for air distribution and the discharge can be made through self-opening hatches.

The basic system is supplemented by forced air cooling of the control element assembly drive motors. Two vane axial fans induce an air flow through the seismic skirt which is located above the reactor pressure vessel and surrounds the lower portions of the motors. Air ducted from the main system and discharged to the region above the skirt ensures that cool air enters this subsystem.

The controls and supervisory instruments are located in the control room. The air inlet and outlet temperatures at the cooling coils, pressure differentials and the air outlet temperatures from the seismic skirt are continuously indicated. The system is normally manually operated but safeguards actuation is automatic (see Section 7.3.2).

The containment air recirculation, cooling and iodine removal system is an engineered safety features system, and hence, is part of the plant's engineered safeguards, as defined in Section 6. For more supporting details of components the Technical Specifications, the CQE Manual, the EEQ Manual, and the Regulatory Guide 1.97 Responses should be consulted.

The system flow diagram is shown in P&ID 11405-M-1. The design data for the four air handling units are shown in Table 6.4-1

The containment air cooler units are placed in service as necessary to maintain containment temperature at the desired level during normal plant operation.

Nuclear Detector Well Cooling System

This system cools the out-of-core neutron detectors, which are located in tubes or wells in the reactor compartment annulus between the lower portion of the reactor vessel and the biological shield, and maintains the shield concrete temperature below 150°F. The system consists of two air handling units and vane axial fans installed in parallel in a closed loop. Air is ducted into the reactor compartment, where it passes up through the wells and past the detectors before recirculation to the air handling units. The closed loop arrangement limits argon-41 contamination of the containment air space. The system flow diagram is shown in P&ID 11405-M-1.

Each air handling unit contains HEPA filters and cooling coils operating on the component cooling water system. The air handling units and fans are each rated at 100 percent system design capability and can be isolated from each other so that one unit and fan normally operate with the other fan in standby. The system design data are shown in Table 9.10-3. A differential pressure gauge is installed across the VA-11 A/B filters to provide a means of determining the condition of each filter.

Table 9.10-3 - "Nuclear Detector Well Cooling System Design Data"

Heat Removal Capacity, Btu/hr	173,000
Flow, CFM	16,000
Air Inlet Temperature at Unit, °F	110
Air Outlet Temperature from Unit, °F	100
Cooling Water Flow, gpm	30
Cooling Water Inlet Temperature, °F	90
Cooling Water Outlet Temperature, °F	101.5

Containment Purge System

The containment purge system was designed to purge the containment by passing up to 50,000 CFM of outside air through the building. The system performs the following functions:

a. Provides means for the reduction of concentrations of radioactive particulates and noble gases in the containment; the latter cannot be reduced by the filtration equipment in the internal containment recirculation system.

b. Ventilates the building to provide a suitable environment during personnel access.

c. Allow the addition of temporary charcoal filtration to the containment purge filter housing as required to reduce iodine releases to the environment.

The purge supply system consists of two flow paths which tie into the containment recirculation ductwork.

The purge exhaust system consists of four flow paths. The two high volume flow paths each consist of an air handling unit which contains a vane axial fan and filter. The two low volume flow paths each include an axial flow fan, recirculation control valve, purge control valve and flow element. The recirculation control valves and purge control valves are used to vary the flow rate. Valves have been provided to isolate the two low volume flow paths from the high volume paths. Instrumentation, in the control room, records the flow through each flow path. Four pneumatically operated butterfly valves provide for containment isolation on the supply and return ducts at either side of the containment penetrations (see Sections 5.9 and 7.3).

The exhaust air is continuously monitored for radioactive contamination before discharge to the atmosphere through the stack (see Section 11.2.3). A bypass duct from the supply system to the monitoring station at the stack permits the system to operate at full flow capacity with a reduced air flow through the containment. In addition, remote-manually operated butterfly dampers are installed in the supply return and bypass ducts to permit modulation of the air flow. These features permit dilution and/or reduction of the containment exhaust air flow prior to environmental dispersal should the activity level in the undiluted full flow containment exhaust dispersal exceed the acceptable limit. The flow diagram is shown in P&ID 11405-M-1.

The air handling units, filter units and fans are located in the auxiliary building. The system controls and supervisory instruments are located in the control room (see Section 7.6). Flows, temperature and pressure differentials are monitored.

The pressure inside the containment will vary as a result of changes in the ambient air temperatures and leakage from air lines and operators inside the containment. The containment pressure relief line is intended to vent the containment when necessary. The flow diagram is shown on P&ID 11405-M-1.

9.10.2.4 Control Room Air Conditioning System

The control room air conditioning system consists of two, air cooled split system package air conditioning units, each rated at 100 percent of the system design capacity, and supply and return ductwork. The system is designed for normal operation at 18,000 CFM total air volume with 1000 CFM of outside ventilating air makeup. The air cooled condensers for each freon refrigeration unit is located on the auxiliary building roof above Room 69. The air condenser units are protected from tornado winds with a windscreen. A waterside economizer utilizing CCW is placed in the 18,000 CFM duct to assist in control room cooling when the CCW temperature is less than 75°F.

The control room HVAC is an essential auxiliary support system and hence, is part of the plant's engineered safeguards, as defined in Section 6. For more supporting detail of components the Technical Specifications, the CQE Manual, the EEQ Manual and Regulatory Guide 1.97 Responses should be consulted.

The system conditions three individually controlled zones, the main control room area, the computer room and the office areas. In addition to the conventional space conditioning in the control room area, a part of the air supply is ducted through the control panels and instrumentation cabinets to provide direct cooling of the enclosed equipment. The system was designed to maintain a space temperature of 78°F at 50 percent maximum relative humidity. A humidifier is installed in the computer room to maintain this area at a constant relative humidity of 50 percent. The flow diagram is shown in P&ID 11405-M-97.

A thermometer is in the control room at all times for monitoring of control room temperature (Reference 9.10-11).

Two HEPA and charcoal filter assemblies (VA-64A and VA-64B), with separate booster fans (VA-63A and VA-63B) rated at 2000 CFM, are installed at the outside air makeup intake to the system. Normally this equipment is bypassed, but if a VIAS is received, one fan is automatically started, the associated filter is brought on line and the unfiltered intake is isolated. The fan that is automatically started is the fan associated with the control room air conditioning unit operating at the time of the event. The other fan and associated filter is automatically placed in standby and will start automatically if a low flow or damper misalignment signal is received. The operating filter unit filters 1000 CFM of outside air and 1000 CFM of recirculation control room air. The control room is maintained at a positive pressure while in this mode ("Filtered Air Makeup Mode") to preclude air infiltration from outside the control room envelope. This sequence will also be automatically initiated by a main steam relief valve open signal. There is a remote possibility for a simultaneous initiation of VIAS and a toxic gas accident during which toxic gas monitors will isolate the control room by closing the fresh air dampers PCV-6681A and PCV-6681B and by shutting down the control room ventilation. The control room ventilation will remain in shutdown mode and will be returned manually to the filtered mode of operation after control room operators have taken protective actions (see Section 14.23). Local differential pressure indication and remote alarms are provided across each filter unit (VA-64A and VA-64B) to provide a means of determining conditions of the filter. Remote air flow and temperature indication and alarm is provided for each unit.

The air conditioning units are located in the southwest corner of the control room. Controls for each unit are located on control room panels AI-106A and AI-106B. Toxic gas monitors at the fresh air intake (VA-65) provide for continuous measurement of fresh air to the control room in order to detect ammonia which may be released during an offsite chemical accident. The toxic gas monitors are designed to detect ammonia. Upon detection of toxic gas beyond the alarm setpoint, the control room will be isolated by automatically shutting the fresh air dampers (PCV-6681A and PCV-6681B) and by automatically shutting down the control room ventilation to preclude the infiltration of toxic gas (see Section 14.23) (References 9.10-3, 9.10-4, and 9.10-10).

9.10.2.5 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 (vol) following the extremely unlikely event of a major loss-of-coolant accident (LOCA) (Ref. 9.10-5).

The containment hydrogen purge system is an engineered safety features system and hence, is part of the plant's engineered safeguards, as defined in Section 6. For more supporting detail of components the Technical Specifications, the CQE Manual, the EEQ Manual and Regulatory Guide 1.97 Responses should be consulted.

The system consists of two purge units, each with its own 250 cfm positive displacement blower, inlet and outlet ducts, isolation valves, and two hydrogen analyzers. The purge system is manually operated and is normally isolated from the containment by locked closed valves.

The hydrogen detection system can sample the containment atmosphere at various levels via six connections. Each sample line is provided with two normally closed, remotely operated valves powered from redundant power sources. The sample is passed to a common manifold header which passes through the containment via redundant mechanical penetrations. The sample is measured with a hydrogen analyzer and returned to the containment via another mechanical penetration.

Hydrogen purge system exhaust is normally routed through HEPA and charcoal filters prior to entering the ventilation discharge duct. A normally isolated filter bypass is available to enable purging operations to continue in the unlikely event that the purge filter fails to pass flow for some reason. Radioactivity discharged by the hydrogen purge system is measured by the stack monitoring system.

9.10.2.6 Radioactive Waste Processing Building HVAC

The Radioactive Waste Processing Building Heating and Ventilation System consists of two 50% capacity supply air handling units, one 100% exhaust filter package, two 50% capacity exhaust fans, associated dampers, accessories and controls. Each of the supply air handling units consists of a fan, filters and a supply heating coil. The ductwork on the downstream side of each air handling unit and fan has a backdraft damper. There is a pneumatic operated isolation damper upstream of each air handling unit, each exhaust fan, and the HEPA filter package. The exhaust air filter package can be isolated by the isolation dampers upstream of the Filter package and the two isolation dampers downstream of the package, upstream of the exhaust Fans. The system supplies filtered heating and ventilation to limit the summer building temperature to 105°F maximum and the winter building temperature to 70°F minimum. Controls for the system are pneumatic and electric.

The supply and exhaust fans are controlled by handswitch's on the local HVAC control panel. In addition to automatic trip on motor overload, the fans are interlocked such that each exhaust fan always leads its respective supply fan on start-up and lags the supply fan on shutdown. Therefore, if an exhaust fan trips, the associated supply fan will also trip, to prevent pressurization of the building. Exhaust from the building passes through HEPA filters and a radiation monitor samples the exhaust before discharge to the atmosphere.

Radioactive Waste Processing Building Office Area HVAC System

The Radioactive Waste Processing Building Office Area HVAC consists of 100% packaged rooftop air conditioner with electric heating coil and filter with supply and return ductwork and ductwork accessories. This system supplies filtered, heated, and cooled air to maintain the office area temperature of 75°F±3° (78°F Summer/72°F Winter). The package roof top unit shall include controls which are electric and integral to the unit by a room thermostat, located in Room 511.

9.10.2.7 Chemistry and Radiation Protection (CARP) Building HVAC Systems

The HVAC for the CARP Building is subdivided into the Laboratory Area, Office Area, OPPD Locker Rooms, and Contractor Locker Rooms HVAC systems. All cooling coils are direct expansion, all heating coils are electric and all condensers are air-cooled. The HVAC systems are designed to maintain the CARP Building at 75 ± 3°F.

The Laboratory Area HVAC System consists of a commercial quality, penthouse, multizone, air handling unit with a remote condenser for supply air, and a Regulatory Guide 1.140 HEPA Filter Package for exhaust air. The Laboratory exhaust ties into the Radioactive Waste Processing Building exhaust where it is monitored prior to release to the atmosphere. The Chemistry Counting Room supply is filtered through a HEPA Filter Package. The areas served by the Laboratory HVAC system are maintained at a negative pressure with respect to adjacent areas in the CARP Building. The Chemistry Counting Room and the Computer Room are maintained at a positive pressure with respect to adjacent areas served by the Laboratory HVAC System.

The Office Area HVAC System consists of a commercial quality, package, roof-top, variable air volume unit. The office area is maintained at a positive pressure with respect to the adjacent Laboratory Area, Locker Room Areas, and Auxiliary Building Areas.

The OPPD and Contractor Locker Rooms HVAC Systems consist of commercial quality, packaged, roof-top, constant volume units. Exhaust fans are provided for the toilet and shower areas. The Locker Room areas are maintained at a negative pressure with respect to the adjacent office areas.

9.10.2.8 Office/Cafeteria Addition HVAC Systems

The HVAC for the Office/Cafeteria Addition is subdivided into the Cafeteria Area and Office Area HVAC systems. All cooling coils are direct expansion, all heating coils are electric, and all condensers are air cooled. The HVAC systems are designed to maintain the Office/Cafeteria Addition at 75 ± 3°F.

The Cafeteria Area HVAC system consists of a commercial quality, single zone, air handling unit with filter and mixing sections, and remote condensing unit for supply air. Exhaust fans are provided for exhausting at dishwashing and cooking/frying areas. The cafeteria area is maintained at a positive pressure with respect to the adjacent CARP Building.

The Office Area HVAC System consists of a commercial quality, single zone, air handling unit with filter and air mixing sections and remote condensing unit for supply air. An exhaust fan is provided to exhaust the toilet and janitor rooms. The Office Area is maintained at a positive pressure with respect to the adjacent CARP Building.

9.10.3 System Components

9.10.3.1 Auxiliary Building Ventilation System

The air handling supply units in the controlled and uncontrolled access area systems are of similar design. The steam heating coils are commercial finned tube units and the roughing filters are the automatically advanced roll type. This equipment is installed in galvanized steel housings with appropriate access provisions.

The supply and exhaust fan data are shown in Table 9.10-4.

Table 9.10-4 - "Auxiliary Building Fan Data"

	Controlled Access Area Supply	Controlled Access Area Exhaust	Uncontrolled Access Area Supply	Uncontrolled Access Area Exhaust
Item No's, VA-	35A&B	40A, B & C	45A&B	41
Number Installed	2	3	2	1
Design Air Flow, CFM per fan	36,250	24,200	13,000	20,000
Motor Rating, HP per fan	60	60	10	20

The controlled access area exhaust filter unit consists of a leak-tight galvanized steel housing containing three HEPA filter compartments each of which can be isolated and is separately accessible. These filters are similar to those used in the containment air recirculation, cooling and iodine removal system as discussed in Section 6.4.3 but differ in that they use replaceable prefilter elements ahead of the HEPA elements as opposed to mist eliminators.

The charcoal filters in the bypass ducts at the exhausts from the emergency safeguards pump and piping compartments and the spent fuel pool area are of pleated design with one inch thick adsorber beds. The activated charcoal is similar to that in the containment air recirculation, cooling and iodine removal system filters as described in Section 6.4.3. The filter design ensures that there is no bypassing around the beds. A differential pressure gauge is installed across each filter to provide a means of determining condition of each filter.

The controlled access area exhaust ductwork is welded and flanged and is leak tight. All other ductwork is of conventional design and fabrication. The controlled access area supply dampers are multi-blade design with blade seals for tight shut off. The exhaust system butterfly valves provide "bubble-tight" shut-off to effect complete compartment exhaust isolation.

The ventilation discharge duct is a cylindrical steel structure located close to, and laterally supported from the containment wall; it terminates close to the top of the containment stressing gallery.

9.10.3.2 Turbine Building Ventilation System

The turbine building main ventilation equipment is of standard, commercial design. The fans are centrifugal machines; data are shown in Table 9.10-5.

Table 9.10-5 - "Turbine Building Fan Data"

	Supply	Exhaust
Item No's VA-	151A thru D	158A thru P
Number Installed	4	14
Air Flow, CFM per fan	125,000	35,715
Static Pressure Rise, in. H_2O	3.25	7.5
Motor Rating, HP per fan	100	7.5

9.10.3.3 Containment Air Cooling and Ventilation Systems

The containment air recirculation system air handling components are discussed in Section 6.4.3. The HEPA filters used in the nuclear detector well cooling system are similar to those of the above system.

The nuclear detector well cooling coils are the standard drainable, finned tube type. The air handling unit housings are of leak tight galvanized steel construction.

The purge supply system air handling units are similar in design to those already described for the auxiliary building systems.

The two purge system exhaust filter units are similar to the auxiliary building controlled access area units.

The nuclear detector well and purge exhaust ductwork is welded and flanged and is leak tight. Elsewhere, ductwork is of conventional design and fabrication except that duct sections in the containment are flanged to permit removal should cleaning and decontamination ever become necessary. The containment purge isolation butterfly valves are designed for a maximum leak rate of 0.01 SCF per hour at the DBA conditions of 60 psig, 288°F and 100 percent humidity.

Containment system fans are pre-set, adjustable pitch, air-over-motor, vane axial machines. The recirculation and cooling system main fans are discussed in Section 6.4.3. Other fan data are shown in Table 9.10-6.

Table 9.10-6 - "Containment Air Cooling Systems, Fan Data"

	CEDM Cooling	Nuclear Detector Well	Purge Supply	Purge Exhaust		
Item No's VA-	2A&B	12A&B	24A&B	32A&B	76	77
Number Installed	2	2	2	2	1	1
Air Flow, CFM per fan	10,000	16,000	25,000	25,000	0-10,000	0-2500
Static Pressure Rise, in. H_2O	3.3	6.9	6.8	6.9	2.5	3.0
Motor Rating, HP per fan	15	40	40	40	7.5	3.0

9.10.3.4 Control Room Air Conditioning System

The air conditioning units are standard machines with hermetic compressors and air cooled condensers. A water side economizer utilizing CCW is required to provide the needed control room cooling when the ambient temperature is below 0°F (Reference Calculation FC06311). Ductwork was designed to commercial standards.

The emergency intake HEPA and charcoal filters are installed in air tight housings normally isolated at either end with butterfly type tight shut-off dampers. The HEPA filters are similar to those previously described for the other systems. The charcoal filters consist of V-bed units containing two inch deep beds of activated charcoal which is similar to that in the containment air recirculation, cooling and iodine removal system filters as described in Section 6.4.3. The filter design ensures that there is no bypassing around the beds. The filter booster fans are centrifugal machines with a capacity of 2000 CFM. A heater is provided in each filter unit to maintain maximum air stream relative humidity below 70%.

9.10.3.5 Radioactive Waste Processing Building HVAC Systems

The make-up air handling units are of the central station packaged type consisting of fan, heater, and filter sections mounted together in a common housing on a common base frame. The exhaust fans are centrifugal type with radial blade wheels mounted on a fan/motor isolation base. The exhaust filter package is designed to the requirements of ANSI N-509, 1980 and provides HEPA Filtration of the building exhaust air flow, which is also monitored.

Table 9.10-7 - "Radioactive Waste Processing Building Fan Data"

	Make-up Air Handling Units	Process Exhaust Fans	Offices HVAC Unit	Toilet Exhaust Fan
Item Number VA	600 A+B	30 A+B	602	604
Number Installed	2	2	1	1
Design Air Flow Rate Per Fan (CFM)	7375	8350	2450	150
Design Total Pressure Rise (Inches H_20)	4.0	16.8	1.0	.25

9.10.3.6 Chemistry and Radiation Protection (CARP) Building HVAC Systems

The Laboratory Area HVAC Supply System consists of one 100% multizone air handling unit, nine electric heating coils mounted in the zone supply ducts, two 50% packaged electronic humidifiers, and a HEPA filter package located in the Chemistry Counting Room supply duct. The air handler consists of a supply fan, filter section, electric heating coil, direct expansion cooling coil, outdoor air intake damper and multizone dampers. The condensing unit consists of two 50% compressors, condensing coils and condenser fans.

The Laboratory Area exhaust system consists of one 100% Regulatory Guide 1.140 filter package, an exhaust fan, inlet and outlet isolation dampers, and welded stainless steel ductwork. The filter package consists of prefilters and HEPA filters.

The Office Area HVAC System consists of one 100% variable air volume air handling unit, a packaged electronic humidifier, and 12 variable air volume terminal control units with electric heat. The air handling unit consists of a fan, filter section, electric heating coil, direct expansion cooling coil, compressor, condenser section, return air and maximum outside air modulation dampers, and exhaust fan section.

The OPPD and Contractor Locker Room HVAC system consist of one 100% constant volume air handling unit and an electric heating coil mounted in the Women's Locker Room supply duct. The air handling unit consists of a fan, filter section, minimum outside air damper, exhaust air, return air and maximum outside air modulating dampers and an exhaust fan section. The systems also include exhaust fans for the toilet and shower areas.

Table 9.10-8 - "Carp Building-Fan Data"

	Laboratory Air Handling Unit	Process Exhaust Fan	Office Area HVAC	Contractor Locker Room HVAC	OPPD Locker Room HVAC
Item Number					
VA	652	651	636	632	628
Number Installed	1	1	1	1	1
Design Air Flowrate (action)	9800	12000	9000	4000	8000
Design Total Pressure Rise (in. H_20)	5	13	2	1	1.74
Fan hp.	20	40	3	3	3
Item Number					
VA	629	630	633	673	634
Number Installed	1	1	1	1	1
Design Air Flowrate	1300	640	945	1880	695
Fan hp.	1/2	1/4	1/3	3/4	1/4

9.10.3.7 Office/Cafeteria Addition HVAC Systems

The Cafeteria Area HVAC System Consists of one 100% constant volume single zone air handling unit, one duct mounted electric reheat coil, one electric propeller type unit heater, one electric baseboard radiator, and two roof mounted exhaust fans. The air handling unit consists of a supply fan and direct expansion cooling coil section, electric heating coil section, filter section, and air mixing section with outdoor and return air dampers. The condensing unit consists of three 33% compressors, condenser coils, and fans.

The Office Area HVAC System consists of one 100% constant volume single zone air handling unit, two duct mounted electric reheat coils, two electric, baseboard radiation and one roof mounted exhaust fan. The air handling unit consists of a supply fan and direct expansion cooling coil section, electric heating coil section, filter section and air mixing section with outdoor and return air dampers. The condensing unit consists of two 50% compressors, condenser coils, and fans.

Table 9.10-9 - "Office/Cafeteria Fan Data"

	Cafeteria HVAC	Office Area HVAC	Restroom Exhaust Fan	Dishwasher Exhaust Fan	Kitchen Exhaust Fan
Item Numbers					
VA	702	703	709	710	711
Number Installed	1	1	1	1	1
Design Air Flowrate	5200	5600	600	650	2500
Design Total Pressure Rise	3.89	3.71	.625	.625	.625
Fan hp	7.5	7.5	1/2	1/2	1/2

9.10.4 System Operation

9.10.4.1 Auxiliary Building Ventilation System

Auxiliary Building Controlled Access Area System

The system operates continuously and once started, normally requires minimal supervision. If high discharge radiation activity is alarmed several options are open to the operator:

a. If the compartment containing the source of activity is known, then this compartment can be individually isolated by closing the dampers;

b. Compartments and areas can be isolated and the dampers sequentially opened to identify the activity source;

c. The system can be operated in conjunction with the containment purge supply and bypass system to reduce the discharge concentrations by dilution.

Should the containment be undergoing a purge and the stack monitor indicates high radioactivity, the radiation monitoring cabinet permits samples of the containment atmosphere and the auxiliary building exhaust to be independently monitored. This allows the building in which the activity source is present to be identified and the appropriate action can be taken.

In the event of leakage in any of the safeguards pump rooms during post-DBA recirculation cooling, high radiation at the exhaust stack might be alarmed. If continued operation with leakage is possible the compartment affected cannot be isolated since the equipment would overheat. In this case the bypass charcoal filter in the compartment exhaust duct is brought on line to pass the exhaust through the filter. This subject is also discussed in Section 6.2.6.3. An EOP/AOP Attachment provides instructions for restoring Auxiliary Building Controlled Area ventilation after an accident. The purpose of that attachment is to restore forced air flow through the SI pump rooms for equipment cooling in the event of a DBA coincident with a loss of offsite power. An emergency key operated bypass switch allows the supply and exhaust fans to remain in operation under off-normal conditions if a spurious fan trip occurred (such as a smoke detector, freezestat, or duct pressure alarm).

Auxiliary Building Uncontrolled Access Area System

The system operates continuously and requires minimal supervision. There are no special operating procedures either during normal operation or in any emergency situation.

9.10.4.2 Turbine Building Ventilation System

This system with exception of the engine driven auxiliary feedwater pump room also operates continuously and requires minimal supervision. There are no special operating procedures either during normal operation or in any emergency situation.

The ventilation system for the engine driven auxiliary pump room operates automatically when the engine operates. When the engine starts, one of the two fans is energized. If the temperature in the room increases to above 105°F, the second fan is energized. If normal electrical power to the fans is unavailable, the fan's load automatically transfers to the integral engine driven auxiliary generator.

9.10.4.3 Containment Air Cooling and Ventilation Systems

Containment Air Recirculating, Cooling and Iodine Removal System

The system operates continuously during reactor operation. It may also be operated prior to and during containment purging, in order to assist in the reduction of radioactive airborne contaminants, and during shutdown periods to supplement the purge system by providing additional air cooling and filtration. This latter mode is not a design requirement however.

At normal design conditions, operation of either both the cooling and filtering units or one of the cooling and filtering units and both cooling units is required as discussed in Section 9.10.2.3. However, during the winter months the cooling system water is at a temperature significantly below the design temperature, having the effect of increasing the cooling capacity of the coil units. Also, the lower ambient outside temperatures result in increased heat losses from the containment. Under these circumstances it is probable that the system could be operated with less than the design number of units on line. However, such partial system operation will only be undertaken once it has been demonstrated that equipment relying upon air cooling, in particular the reactor coolant pump motors, are not starved of air.

The stators of these motors are instrumented to provide temperature indication with appropriate high temperature alarms.

Nuclear Detector Well Cooling System

The system operates continuously during reactor operation with one air handling unit in operation and the other on standby. During reactor shutdown periods the system is normally shut down but it can be used to provide ventilation when maintenance is being carried out in the nuclear detector well area.

Containment Purge System

The purge system is operated under the following circumstances:

a. During reactor shutdown periods;

b. In the containment bypass mode, to provide dilution of the auxiliary building controlled access area ventilation exhaust, should this ever be required.

If the system is operating and the emergency safeguards are initiated or the discharge duct monitor indicates high radioactivity, the purge system duct dampers at the containment penetrations are automatically closed and the fan motors are tripped (See Sections 7.3.2.6 and 11.2.3.2) (References 9.10-6, 9.10-7 and 9.10-8).

9.10.4.4 Control Room Air-Conditioning System

The system normally operates continuously with one air-conditioning unit operating and the other on standby, but both units can operate simultaneously if heat load dictates. A VIAS will allow up to two stages of the three stage refrigeration compressor to operate, to limit electric diesel generator loading. There is an override manual control on this VIAS to allow three stage compressor operation. VIAS will also close the CCW inlet and outlet valves on the economizers. There is one mode of operation at normal plant conditions:

a. With 1000 CFM of outside make-up, 17,000 CFM recirculated and the cooling coils operable. The system is under automatic control from the space thermostats. The air intake HEPA and charcoal filter units are bypassed.

   This mode of operation can be initiated from the control room. A selector switch for each unit also provides for the following operating conditions.

b. As in (a) above except that one of the HEPA and charcoal filter units is brought on-line, the associated booster fan is started and the normal air intake is closed. The second filter unit is maintained in a standby condition. The toilet exhaust fan is tripped and the associated damper is closed.

c. 18,000 CFM of air is internally circulated with all outside dampers closed. The system is under automatic control from the space thermostats.

d. All outside dampers are closed and the control room ventilation is shutdown to preclude the infiltration of ammonia toxic gas during a toxic gas release accident.

Condition (b) is the design mode of operation in the event of the DBA. The system is automatically brought to this condition by the ventilation isolation actuation signal (see Section 7.3.2.6). Following a DBA, it is still possible to select condition (c) by operation of the mode selector switch.

Condition (d) is the design mode of operation in the event of a toxic gas release accident. The system is automatically brought to this condition via the alarm signal from the ammonia toxic gas monitors. Following a toxic gas accident it is still possible to select position (a), (b), or (c) by using the toxic gas isolation override switch and the mode selector switch for the control room ventilation systems (see Section 14.23).

9.10.4.5 Radioactive Waste Processing Building HVAC Systems

This system operates continuously and requires minimal supervision. Control system design provides interlocking to prevent operation of building supply system without associated exhaust fans on. The system is designed to allow operation at 50% capacity to allow for energy savings during periods of low building use.

9.10.4.6 Chemistry and Radiation Protection (CARP) Building HVAC Systems

The CARP Building HVAC Systems operate continuously and require minimal supervision. Controls for the systems are electric/ electronic.

9.10.4.7 Office/Cafeteria Addition HVAC Systems

The Office/Cafeteria Addition HVAC Systems operate continuously and require minimal supervision. Controls for the systems are electric/ electronic.

9.10.5 Design Evaluation

The heating, ventilation and air-conditioning systems provide a suitable environment for equipment and personnel over a design range of ambient outside temperatures from -11°F to 95°F (dry bulb) and 78°F (wet bulb). In those areas where airborne radioactivity could constitute a hazard, the reduction of this activity is the prime design consideration. Administrative procedures ensure that doses to personnel in these areas are within 10 CFR Part 20 limits during normal plant operation. The systems have the capability of limiting off-site release of airborne contaminants to concentrations below those specified in 10 CFR Part 20 by filtration, dilution and, if necessary, isolation.

The auxiliary building, containment and control room systems are designed to survive the seismic loadings imposed during the maximum hypothetical earthquake without damage or any change or loss of function.

9.10.6 Availability and Reliability

The heating, ventilating and air-conditioning equipment is, as a minimum, in accordance with accepted industrial standards and was designed to operate continuously for extended periods without attention.

The nuclear detector well and control room systems are provided with 100 percent standby capacity. The containment air recirculation, cooling and iodine removal system is a multi-unit arrangement with 50 percent excess installed capacity, and since it is also an emergency safeguards system (see Section 6.4), it was designed and fabricated to standards which reflect the importance of the safeguards function. Where standby equipment is provided, operation will be on an alternating basis to provide assurance of operability.

The purge, auxiliary and turbine building systems have multiple fans so that in the event of fan failure, the systems can still function but at reduced capacity.

The auxiliary building systems are shut down during the short and infrequent periods when the intake filter rolls are replaced. The arrangement of the auxiliary building controlled access area exhaust HEPA filters permits one-third of the filtration capacity to be replaced without disturbing the remainder. The system operates at slightly reduced capacity during filter replacement. Purge system filter replacement can be scheduled for the periods when this system is shut down.

The systems are dependent upon the electrical and component cooling water systems except for control room HVAC refrigeration system which are discussed in Sections 8 and 9.7 respectively. (The control room HVAC system utilizes CCW for economizer coils only.)

The control element drive mechanism motors are cooled by a combination of the main air handling units and the CEDM cooling fans. The former deliver cooled air to the area of the drive motors and the latter ensure that air is withdrawn from the drive motor region downwards through the seismic skirt. The function of the CEDM cooling fans is to remove heat from the drive housings, thus lowering temperatures inside the mechanisms and reducing wear, and to keep instrumentation cables cool. Neither function is necessary for the continued short term operation of the plant. There is concern over mixing of the cool air being supplied to the drive motors and hot air rising from the housings when the fans are not operating.

Continued short term operation of the plant, so far as cooling of the drive motors is concerned, depends only on availability of the main air handling units. These units must be in operation to cool other loads in the plant, e.g., reactor coolant pump motors. Continued operation of the plant without any main air handling units is prohibited for multiple reasons. The main air handling units are very reliable units and redundancy is available.

The neutron detector well cooling units serve the following two functions:

a. Maintain concrete temperatures in the concrete shielding surrounding the reactor vessel below 150°F.

b. Maintain neutron detector temperature below 300°F.

The neutron detector cooling system comprises two independent cooling units, either of which can handle the required cooling load. It is considered unlikely that both would be out of service at the same time.

Thermal detectors are buried at strategic locations in the concrete. If both cooling units should be out of service simultaneously, the operator can use this instrumentation to evaluate how rapidly the reactor must be cooled down to avoid concrete damage. Since the neutron flux is high at the location of the sensors, a loss of cooling units test was run early in life to evaluate the required rapidity of shutdown. This test was run after criticality since the major source of heat in the concrete is radiation absorption.

If concrete is heated above 150°F, it is likely to suffer a reduction in strength. The magnitude of the reduction would be dependent on the rate of temperature rise, the maximum temperature attained and the duration of the elevated temperature. The limit of 150°F is consistent with the requirements of ACI-318-63.

This loss of cooling unit test provided data to evaluate temperature at the neutron instrumentation. Permanent temperature measuring devices were not installed at this location since the neutron flux is too high. The neutron instruments have been shown by test to be suitable for temperatures of 300°F and less. Above this level, the signal deteriorates.

The Radioactive Waste Processing Building Heating and Ventilating System has two 50% capacity exhaust fans and two 50% capacity supply fans so that the system can still operate at reduced capacity in the event of a fan failure.

9.10.7 Tests and Inspections

All HEPA and charcoal filters were subject to performance tests prior to delivery. Testing procedures followed those outlined in Section 6.4.7. Heating and cooling coils were pressure tested at the manufacturer's shop.

All leak-tight ductwork and equipment housings were pressure tested with air at 0.5 psig after installation and all joints were examined for leaks using soap solution. The testing of the purge system containment isolation valves is discussed in Section 5.9.6.

After installation the system was tested with regard to flow paths, flow capacities, heating and cooling capabilities, mechanical operability and filtration efficiency. Dampers and the pumps and valves of associated systems were tested for operation at the proper setpoints. Controls, instruments and alarms were checked for operability and adequacy of limits.

The testing and inspection of the containment air recirculation, cooling and iodine removal system equipment is discussed in Section 6.4.7. The HEPA and charcoal filters in the auxiliary building, purge and nuclear detector well systems are periodically inspected or replaced. The control room filters can be tested or replaced during plant operation.

Equipment outside the containment is accessible for test and inspection during plant operation. Access to the containment is restricted during reactor operation, but is sufficient to allow for the inspection of key components.

9.10.8 Specific References

9.10-1 Safety Evaluation Report Supporting Amendment Number 52 to the FCS Technical Specifications, October 14, 1980

9.10-2 Deleted

9.10-3 Safety Evaluation Report Supporting Amendment Number 87 to the FCS Technical Specifications, April 29, 1985

9.10-4 Safety Evaluation Report Supporting Amendment Number 107 to the FCS Technical Specifications, March 30, 1987

9.10-5 Safety Evaluation Report Supporting Amendment Number 138 to the FCS Technical Specifications, March 19, 1991

9.10-6 Safety Evaluation Report Supporting Amendment Number 68 to the FCS Technical Specifications, February 24, 1983

9.10-7 Safety Evaluation Report Related to the Mechanical Operability of Purge/Vent Valves, February 24, 1982 (Cartridge 923, Frame 1771)

9.10-8 Safety Evaluation Report Related to Minimum Containment Pressure Setpoint, July 28, 1981 (Cartridge 997, Frame 1823)

9.10-9 Safety Evaluation Report Supporting Amendment Number 154 to the FCS Technical Specifications, August 10, 1993

9.10-10 Safety Evaluation Report Supporting Amendment Number 183 to the FCS Technical Specifications, January 28, 1998

9.10-11 NRC Amendment 188 to Technical Specifications, December 31, 1998.

9.10.9 General References

9.10.9.1 Safety Evaluation Report Supporting Amendment Number 128 to the FCS Technical Specifications, April 12, 1990

9.10.9.2 Safety Evaluation Report Supporting Amendment Number 15 to the FCS Technical Specifications, September 3, 1976

9.10.9.3 NRC Generic Letter 92-16, NUREG-0737 Related Items, October 27, 1982

9.10.9.4 NRC IE Bulletin Number 80-03, Loss of Charcoal from Standard Type II Tray Absorber Cells, February 6, 1980

9.10.9.5 Safety Evaluation Report Supporting Amendment Number 67 to the FCS Technical Specifications, February 2, 1990

9.10.9.6 Safety Evaluation Report Supporting Amendment Number 130 to the FCS Technical Specifications, May 23, 1983

9.12 COMPRESSED AIR SYSTEM

9.12.1 Design Bases

The compressed air system provides compressed air to the instrument air and the service air headers.

The instrument air header provides air for pneumatic controls and the actuation of valves, dampers, and similiar devices, as well as the fuel handling machine in the containment. The service air system provides air for portable maintenance tools (such as stud tensioners and chipping hammers), demineralizer resin bed mixing, large valves, and the air operated hoist in the intake structure.

The system has the following design bases:

a. Instrument and service air regulation between 80 and 100 psig;

b. Maximum instrument air dew point of -20°F;

c. Quantity of instrument air sufficient to provide for all pneumatic controls and valve and damper operators expected to operate under normal and post accident operating conditions;

d. Quantity of service air sufficient for all equipment expected to operate simultaneously plus an allowance for the use of maintenance tools;

e. Failure of the service air distribution system not to cause the loss of instrument air.

System operation is not required to initiate operation of engineered safeguards equipment since all air operated valves and dampers required to control the accident were designed to assume the accident-controlling position on loss of air pressure or are provided with safety grade passive accumulators or nitrogen backup systems. Containment penetrations for the hydrogen purge system and hydrogen analyzer VA-81B have fail-open, air-operated valves which receive an auto-close signal on CIAS, but these valves do not have backup accumulators. Acceptability of this design is discussed in Section 5.9.5. Further, air is not required for the reactor protective system.

Those portions of the instrument air system that are required to operate engineered safety features or essential auxiliary support systems valves, e.g., from the check valves downstream through the accumulators, tubing and components, are part of the plant's engineered safeguards, as defined in Section 6. The instrument air system is designed in accordance with the requirements of Appendix N. See P&IDs 11405-M-264. For more supporting detail of components, the Technical Specifications, the CQE Manual, the EEQ Manual, and the Regulatory Guide 1.97 Responses should be consulted.

9.12.2 System Description

The compressed air system is shown in P&ID 11405-M-263. The major components of this system are located in the basement of the auxiliary building on the east side.

Air is supplied by three identical two stage compressors which operate automatically to maintain air pressure. The compressors are connected to a discharge manifold which feeds the instrument and service air systems.

Instrument air flows first through an air receiver which holds a reserve supply of air. From the receiver it flows through a prefilter, an air dryer, an afterfilter, and then to the distribution system.

A side stream dewpoint analyzer continuously monitors the dewpoint of the instrument air supply downstream of the air dryers.

The service air also flows first to a receiver. If the pressure in the instrument air system drops below a set level, a pressure control valve down stream of the receiver closes, shutting off flow to the service air system.

9.12.3 System Components

The air compressors are two stage, intercooled and aftercooled, heavy duty type with non-lubricated cylinders. The unit capacity is 710 SCFM oil-free, dry air at 125 psig. The intercoolers, aftercoolers and cylinder jackets are cooled by the turbine plant cooling water system.

The receivers are vertical steel tanks designed and fabricated in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.

The normal service and standby air dryers are of the desiccant non-heat, regenerated dual tower unit. One tower is in service while the other is regenerated.

Piping is in accordance with USAS B31.1. Instrument air piping is mostly copper. (Reference: Technical Services Analysis Request 87-04)

9.12.4 System Operation

The air supply system operates automatically. When air demand is low, one compressor operates. If the air pressure continues to drop, a second compressor starts. The third compressor is kept on standby and can be operated manually if necessary. The normal instrument air dryer operates and regenerates automatically. The standby dryer also operates and regenerates automatically. The standby dryer is manually placed in service whenever the normal service air dryer is unavailable.

If the instrument air dew point is greater than or equal to -20°F or if there is an indication of high air moisture, instructions are provided to operators for specific actions to correct the situation.

9.12.5 Design Evaluation

The system provides sufficient instrument and service air to satisfy all pneumatic instruments, controls, valves, dampers and other equipment. The compressor capacity provided is in excess of the design maximum demand. Loss of the system would not prevent placing the plant in a shutdown condition from normal plant operation or any accident situation since the instrument air system is not required for a safe plant shutdown.

All air operated valves which are required to operate during loss of instrument air do so by spring actuation following removal of air pressure from their operators, or have air storage tanks (accumulators). The removal of air pressure is actuated by solenoid valves.

Following the trip, decay heat can be removed from the plant through safety valves on the main steam lines. These valves depend only on springs for actuation. The steam removed from the steam generators through the safety valves can be replaced using the motor driven auxiliary feedwater pump.

The instrument air system is also not required for any of the engineered safety features to operate properly. Air operated valves fail upon loss of air pressure in a position for the engineered safeguards to function properly, or have accumulators to keep safety related components operable for a required time. Containment penetrations for the hydrogen purge system and hydrogen analyzer VA-81B have fail-open, air-operated valves which receive an auto-close signal on CIAS, but these valves do not have backup accumulators. Acceptability of this design is discussed in Section 5.9.5. The twelve safety injection valves inside the containment which must be operated after the accident to adjust the flow are electric motor operated.

The functional requirements for safety related components that have accumulators have been established, and functional testing has verified the operability of these components. To make these components independent of the instrument air system during a Design Basis Accident, each is equipped with an air storage tank (accumulator) which is pressurized during normal operations. If the pressure in the instrument air system drops, a check valve in the line to the air storage tank closes, which isolates the tank from the system and provides a supply of air to the component for the required time period. Valves HCV-344, HCV-345, LCV-383-1, LCV-383-2, HCV-400A/B/D, HCV-401A/B/D, HCV-402A/B/D, HCV-403A/B/D, HCV-480, HCV-481, HCV-484, HCV-485, HCV-438B and HCV-438D do not depend on an air storage tank (accumulator) upon loss of Instrument Air; dedicated N_2 cylinders are interfaced to these valves. Table 9.12-1 includes the time requirements for accumulator operability, and valve operability requirements following a design basis event.

9.12.6 Availability and Reliability

The system was designed for a high standard of reliability. Since some of the loads on the system are of an occasional nature (e.g., waste disposal demineralizer service and maintenance tool operations) the system operates at a very low load factor. One air compressor is a spare; operation of one of the others is normally sufficient. Bypasses are provided so that either of the air receivers can be taken out of service. The standby air dryer provides full capacity backup to the normal service air dryer. In the unlikely event neither air dryer is available both air dryers can be bypassed. Segments of both air supply systems can be isolated for maintenance or repair without shutting down the entire system.

9.12.7 Tests and Inspection

The air receivers were hydrostatically tested in the shop. Other components were tested during the functional tests prior to plant operation. The system was tested with regard to flow paths, capacity and mechanical operability. The compressors and valves were tested for actuation at the design setpoints. Pressure and temperature indicating and controlling instruments were calibrated and checked for operability.

The equipment is accessible for inspection and maintenance at all times.

Table 9.12-1 - "Safety Related Valves and Bubblers Operable after Loss of Instrument Air"

Valve Tag	System	Location Of Valve	Length of Time Valve Remains Operable (Hrs) (Note 1)	Fail Safe Position
HCV-238	CVCS	CONT	25.0	OPEN
HCV-239	CVCS	CONT	25.0	OPEN
HCV-240	CVCS	CONT	25.0	CLOSED
HCV-304	HPSI	RM 21	--	OPEN
HCV-305	HPSI	RM 21	--	OPEN
HCV-306	HPSI	RM 13	--	OPEN
HCV-307	HPSI	RM 13	--	OPEN
HCV-385	SI&CS	CORR 4	13.0	OPEN
HCV-386	SI&CS	CORR 4	13.0	OPEN
HCV-400A/B/D	CCW	RM 69	--	NOTE 3
HCV-401A/B/D	CCW	RM 69	--	NOTE 3
HCV-402A/B/D	CCW	RM 69	--	NOTE 3
HCV-403A/B/D	CCW	RM 69	--	NOTE 3
HCV-438B/D	CCW	RM 13	N_2 B/U	OPEN
HCV-480	CCW	CORR 4	24.0 (NOTE 4)	OPEN
HCV-481	CCW	CORR 4	24.0 (NOTE 4)	OPEN
HCV-484	CCW	CORR 4	24.0 (NOTE 4)	OPEN
HCV-485	CCW	CORR 4	24.0 (NOTE 4)	OPEN
HCV-2987	SI	RM 13	24.0	NOTE 3
CV-383-1	SI	RM 21	N_2 B/U	OPEN
CV-383-2	SI	RM 21	N_2 B/U	OPEN
YCV-1045A/B	MS	RM 81	0.5	OPEN
A,B,C,D/FIC-383	SI&CS	RM 25A	12.0	--
PCV-6680A-1	VA	RM 81	8 (NOTE 2)	OPEN
PCV-6680A-2	VA	RM 81	8 (NOTE 2)	OPEN
PCV-6680B-1	VA	RM 81	8 (NOTE 2)	OPEN
PCV-6680B-2	VA	RM 81	8 (NOTE 2)	OPEN
PCV-6682	VA	CR	8 (NOTE 2)	OPEN
HCV-1107A	AFW	CONT	8	OPEN
HCV-1107B	AFW	RM 81	8	OPEN
HCV-1108A	AFW	CONT	8	OPEN
HCV-1108B	AFW	RM 81	8	OPEN
FCV-1368	AFW	RM 19	8	OPEN
FCV-1369	AFW	RM 19	8	OPEN
HCV-2898A	CCW	RM 81	8	CLOSED
HCV-2898B	CCW	RM 81	8	CLOSED
HCV-2899A	CCW	RM 81	8	CLOSED
HCV-2899B	CCW	RM 81	8	CLOSED

Table 9.12-1 - (Continued)

NOTES:

1. According to OSAR 87-10

2. Per MR-FC-87-20

3. These valves fail open on a loss of DC control power to their air solenoids, and fail as-is on a loss of instrument air.

4. These valves are equipped with backup nitrogen accumulators to keep them closed during the injection phase (i.e., pre-RAS period) of a large break LOCA in spite of a loss of instrument air pressure. The duration listed is bounding for the pre-RAS period of a large break LOCA.

9.12.8 General References

9.12.8.1 Operations Support Analysis (OSAR) 87-10, Determine Which Valves with Air Accumulators are Required for Safe Shutdown, April 6, 1988.

9.12.8.2 Letter LIC-89-0098, Response to NRC Generic Letter 88-14, Instrument Air Supply System Problems Affecting Safety Related Equipment, February 21, 1989.

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

10.2 SYSTEM DESIGN AND OPERATION

10.2.1 Steam System

The steam system is shown in P&ID 11405-M-252. Steam from the two steam generators supplies the turbine, which is provided with four moisture separators and six extraction stages for feedwater heating.

Steam generated in the steam generators passes through two lines to a header and then through four lines to the turbine stop valves. Each steam generator is provided with spring-loaded safety valves upstream of the main steam isolation valves; the safety valves (discussed in Section 4.3) discharge to the atmosphere.

The steam dump and bypass system includes five automatically actuated turbine bypass valves to the main condenser. Four valves serve as dump valves and one serves as a bypass valve. The steam flow is regulated by the dump and bypass valves in response to signals received from the average reactor coolant temperature and steam system pressure detectors. The capacity of the steam dump and bypass system is sufficient to prevent lifting of the safety valves following a turbine and reactor trip at full load, and for subsequent removal and dissipation of reactor decay heat (see Section 7.4).

Process and heating system steam is normally supplied from the sixth stage turbine extraction; the plant auxiliary boiler is used when extraction steam is not available.

10.2.2 Condensate and Feedwater System

The condensate and feedwater system is shown in P&ID's 11405-M-253 and 11405-M-254. The feedwater cycle is closed; deaeration is accomplished in the main condensers. Condensate from the hotwells is pumped by two of three 50 percent capacity, electric motor driven condensate pumps through the hydrogen coolers, the generator stator coolers and the steam packing exhauster, then through the five stages of low-pressure feedwater heating to the suction of three 50 percent capacity, electric motor driven steam generator feedwater pumps. Condensate can be demineralized using portable demineralization during plant start-up. The feedwater is pumped through one stage of high-pressure feedwater heating to the steam generators. Each steam generator is provided with a three-element feedwater controller and feedwater regulator valve.

An electric motor driven auxiliary feed pump, a turbine driven auxiliary feed pump or a diesel engine driven auxiliary feed pump can provide sufficient feedwater to the steam generators for removal of reactor decay heat in the event the main feedwater equipment is not available. (See Section 9.4)

The two surface condensers are of the deaerating type, sized to condense the turbine full-load exhaust steam. They are also capable of condensing steam bypassed directly to the condensers from the main steam headers following a turbine trip.

The condensers are of the two-pass, divided water box type operating at the same pressure; each is located under a turbine low-pressure cylinder. The unit is capable of operation at reduced load with one of the condensers out of service. The design of the hotwell level control system is based on the concept of allowing the hotwell condensate level to rise and fall. The hotwells act as surge vessels rather than constant level storage tanks and they absorb system thermal swell due to startup and load changes. The hotwells have sufficient storage capacity for three minutes operation at maximum throttle flow with an equal free volume to accommodate surges. Additional swells caused by fast load changes can be handled by the condensate storage tank should the hotwell capacity be exceeded.

The hotwells are connected through the hotwell equalizer cross-tie which is equipped with an isolation valve. A steam cross-tie between the condenser necks ensures essentially equal operating pressures in the two condensers.

Condenser vacuum is established and then maintained by three mechanical vacuum pumps. Vacuum pump off-gases are monitored for radioactivity and discharged to atmosphere.

The feedwater heaters are arranged in two parallel trains. Each train carries half of the feedwater flow and contains five low-pressure heaters and one high-pressure heater. The heaters are of the closed, horizontal, U-tube type. The three lowest-stage heaters are housed in the condenser exhaust necks. Eighty percent of the full load condensate can be passed through one train of heaters if shutdown of a heater train is required. Extraction steam flow to the feedwater heaters from the turbine is not regulated. Automatic-tripping, non-return valves are provided in the extraction lines to assist in preventing turbine overspeed on loss of load.

The arrangement of the heater drains is shown in P&ID 11405-M-255. Drains from the high-pressure heaters are cascaded to the second and third stage extraction low-pressure heaters and finally to a heater drain tank. Separator drains are also discharged to this tank. The water collected is pumped forward into the condensate line by two of three 50 percent capacity heater drain pumps. The water level in the heater drain tank is automatically controlled by modulating the discharge from the heater drain pumps. An emergency drain line to one of the condenser hotwells is provided to accommodate abnormal transients. Shell drains from the remaining low-pressure heaters are cascaded to the next lower pressure heater and ultimately to the condenser. All heaters, except heaters 4A and 4B, have emergency drain lines to the condenser to prevent heater flooding which could result from abnormal transients or heater drain valve malfunctions.

Chemicals are added to the feedwater upstream of the steam generator feedwater pumps for oxygen scavenging and pH control. Boric acid is added to the secondary side of the steam generator. The condensate chemical feed system is used to inject a boric acid solution into the feedwater just before it enters the steam generator. Boric acid has been found to mitigate (caustic) innergranular stress corrosion cracking (IGSCC), innergranular attack (IGA), and tube denting. When a crack is initiated before boric acid treatment, boric acid will slow wall rupture, but will not delay it indefinitely.

A low power soak (approximately 50PPM boron at 30% power) after refueling provides transport of boric acid into tube - support crevices without introducing significant hideout contaminants. Usually 5 ppm boron is maintained during full power operation.

Failure of a condensate pump automatically starts the standby condensate pump which comes up to speed in approximately 8 seconds; the condensate pumps are designed to operate with cavitation. This transient does not require manual throttling of the feedwater valves or tripping a feedwater pump. Failure of a heater drain pump automatically starts the standby heater drain pump. The heater drain tank level is held within set limits by the emergency drain valve until the standby heater drain pump is up to speed. The drain valve can accommodate the full capacity of one drain pump and discharges the drains directly to the condenser. Failure of a feedwater pump automatically starts the standby feedwater pump. The water storage capacity of the steam generators can provide sufficient feedwater until the standby pump is up to speed.

Normally, the main feedwater and condensate system is used to remove decay heat immediately after power operation if outside electric power is available. If outside power is not available, or if operating conditions require condenser shutdown, the auxiliary feedwater pumps will be used to remove decay heat.

10.2.3 Circulating Water System

Circulating water for the condenser is taken from the Missouri River, pumped through the condenser tubes, and discharged to the river as shown in P&ID 11405-M-257. Three circulating water pumps are provided in the intake structure. Operation of two pumps is sufficient when the river temperature is below 60°F. Provision is made to control surface and frazil ice by recirculating the discharge water.

Motor driven traveling screens are installed ahead of the pump suction. The debris and refuse pick up by the screens are removed by water supplied by one of two motor driven screen wash pumps which normally take suction from the Intake Tunnel. An alternate suction line from the circulating water pumps suction cells, downstream of the traveling screens is also available and generally used during plant startup.

10.2.4 Turbine-Generator

The turbine is an 1800 rpm, tandem-compound, non-reheat unit with one high-pressure and two double-flow low-pressure cylinders with 38-inch last-stage buckets. Saturated steam is supplied to the turbine throttle from the steam generators through four stop valves and four governing control valves. The steam flows through the high-pressure turbine and then through four moisture separators in parallel to two double-flow, low-pressure turbines each of which exhausts to a condenser. Intercept valves are installed downstream of the moisture separators. The heat balance is shown in Figure 10.2-1.

Turbine control is accomplished with a rapid response electrohydraulic control system with the following main components, (see Section 7.4 for system operation):

a. Solid state controller and operation panel.

   1. Speed control unit: This develops the speed error signal by comparing ordered speed with actual speed.

2. Load control unit: This develops the load error signal by comparing a load limit with actual load.

These signals are combined to make a total error signal. The control valves are positioned as required to maintain the ordered speed and load.

b. Steam valve actuating system converts the error signal into the required valve positions.

   1. The main stop valves start and initially load the turbine. One stop valve has an internal by-pass which is used for warming the valve bodies and equalizing pressure. The by-pass is positioned by a servo-amplifier from the warming control selector. The other three stop valves, which are single disc valves, are interlocked so that they can only be opened when the first stop valve is fully open.

   2. The control valves which adjust speed and/or load are administratively operated;

   3. The intercept valves isolate the low-pressure turbines from the moisture separators to prevent overspeed on loss of load.

c. High pressure oil supply system: This is a high-pressure hydraulic system employing fire resistant fluid and is used to actuate the steam valves. The system is independent of the bearing oil system and employs two full capacity, high pressure pumps.

d. Emergency trip or protection system: This monitors the operation of the turbine generator and acts to trip the unit by removing hydraulic system pressure from all control and stop valves, causing them to shut, should a condition occur that could damage the unit.

e. Electric power supply system: These electric power supplies are electronic circuits that transform either 110-Volt ac line power or 420 Hz ac power from a permanent magnet generator into accurately regulated dc voltages necessary to operate the control and logic circuits.

Turbine speed is controlled by a speed governor and an emergency governor; thus there are two separate methods of protection against turbine overspeed. The master trip solenoid valve is provided for remote manual tripping and for automatic tripping in response to electric trip signals which energize the master trip relay. In addition, the Fort Calhoun turbine control system includes the following speed control devices:

1. A speed control system capable of controlling speed during start-up operation and limiting overspeed following 100% load rejection to less than 10% by means of the control valves and intercept valves. Turbine speed during normal (synchronized) operation is a function of system frequency.

2. A mechanically actuated overspeed protective system which trips the main stop valves upon 10% (nominal) overspeed.

3. An electronically actuated overspeed protective system which trips the main stop valves upon 12% (nominal) overspeed.

The above systems provide an independent and redundant means of protecting the turbine against overspeed. Capability for online testing of the overspeed protection system has been provided where it is practical to accomplish the tests without tripping the turbine.

The turbine lubricating oil system supplies oil for lubricating the bearings. A bypass stream of turbine lubricating oil flows continuously through an oil conditioner to remove water and other impurities. The turbine lube oil system consists of the pumps shown in Table 10.2-1. Oil is cooled by either of two coolers.

Table 10.2-1 - "Turbine Lube Oil Pumps"

Pump	Type	Drive	Capacity (gpm)	Discharge Pressure (psig)
Main Pump	Centrifugal	Turbine Shaft	1075	200
Booster Pump	Centrifugal	Oil Turbine	--	15
Motor Suction Pump	Centrifugal	ac Motor	1350	30
Turning Gear Pump	Centrifugal	ac Motor	875	40-45
Emergency Oil Pump	Centrifugal	dc Motor	725	40

The generator, directly driven by the turbine, is a three-phase, 60 Hz, 22,000-Volt, synchronous machine rated at 590,800 kVA at 0.85 power factor, 45 psig H_2 and 0.58 short circuit ratio. The generator stator is conductor cooled using water in a closed cooling circuit and the rotor is hydrogen cooled. The generator has sufficient capability to develop the gross kilowatt output of the steam turbine operating with the control valves wide open at rated steam conditions. Field excitation is provided by an Alterrex Excitation System which converts the ac output of an alternator driven by the main generator shaft to direct current by means of stationary rectifiers. This system eliminates commutator brushes and allows diode maintenance while under load.

10.2.5 Condenser Evacuation System

This system is shown in P&ID 11405-M-261. The condenser vent to the atmosphere is provided with a direct path to the atmosphere for use during condenser startup. This path is used for starting up the plant when vacuum is initially established in the condenser. This path to the atmosphere is monitored by a noble gas monitor.

During normal operation, the condenser evacuation system vents to the atmosphere.