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LIST OF ACRONYMS

AFFF	Aqueous Film Forming Foam
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	Boiler and Pressure Vessel
BGE	Baltimore Gas and Electric Company
CCNPP	Calvert Cliffs Nuclear Power Plant
CC	Component Cooling
CDF	Core Damage Frequency
CE	Combustion Engineering, Inc.
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CFR	Code of Federal Regulations
CSS	Containment Spray System
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
DG	Diesel Generator
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
FCR	Facility Change Request
FSAR	Final Safety Analysis Report
GDC	General Design Criterion
HEPA	High Efficiency Particulate Air
HPSI	High Pressure Safety Injection
HVAC	Heating, Ventilation, and Air Conditioning
ICI	Incore Instrumentation
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LPSI	Low Pressure Safety Injection
MOV	Motor-Operated Valve
MWPS	Miscellaneous Waste Processing System
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NEMA	National Electrical Manufacturers Association
NEOP	Nuclear Engineering Operator Procedure
NFPA	National Fire Protection Association
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSCA	Nuclear Safety Capability Assessment
PASS	Post Accident Sampling System
QC	Quality Control
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RTD	Resistance Temperature Detector

LIST OF ACRONYMS

RWT	Refueling Water Tank
SDC	Shutdown Cooling
SFHM	Spent Fuel Handling Machine
SFP	Spent Fuel Pool
SFPC	Spent Fuel Pool Cooling
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SRW	Service Water
SSC	Structure, System, and Component
SWAC	Saltwater Air Compressors
TEMA	Tubular Exchanger Manufacturers Association
VAP	Value Added Pellet
VCT	Volume Control Tank
WPS	Waste Processing System
ZrB_2	Zirc Diboride

9.0 AUXILIARY SYSTEMS

A legend for the figures in Chapter 9 is located on Figure 9-1.

9.1 CHEMICAL AND VOLUME CONTROL SYSTEM

9.1.1 DESIGN BASIS

The Chemical and Volume Control System (CVCS) is designed to perform the following functions:

a. Maintain reactor coolant activity at the desired level by removing corrosion and fission products;

b. Inject chemicals into the Reactor Coolant System (RCS) to control coolant chemistry and minimize corrosion;

c. Control the reactor coolant volume by compensating for coolant contraction or expansion resulting from changes in reactor coolant temperature and other coolant losses or additions;

d. Provide means for transferring fluids to the radioactive Waste Processing System (WPS);

e. Inject concentrated boric acid into the RCS upon a safety injection actuation signal (SIAS);

f. Control the reactor coolant boric acid concentration;

g. Provide auxiliary pressurizer spray for operator control of RCS pressure during shutdown;

h. Provide a means for functionally testing the check valves which isolate the Safety Injection (SI) System from the RCS (Although this is a design function of the CVCS, these check valves are functionally tested in accordance with the Inservice Test Program.), and for hydrostatic and leak testing of the RCS;

i. Provide continuous on-line measurement of reactor coolant fission product activity.

Portions of the letdown system are American Society of Mechanical Engineers (ASME) Class 1, and thus require a fatigue analysis of the applicable thermal shock transients, and other operational cycles. In addition to design cyclic transients a, d, and e of Section 4.1.1, the letdown system fatigue analysis considers 200 events where letdown flow is lost for an extended period of time. After the letdown piping has cooled to ambient temperature, a restart of letdown flow results in a rapid increase to RCS temperature. See Reference 1 for further details.

Portions of the charging system are ASME Class 1, and thus require a fatigue analysis of the applicable thermal shock transients, and other operational cycles. In addition to design cyclic transients a, b, c, d, e, and g of Section 4.1.1, the charging system fatigue analysis considers 200 loss of letdown events and 200 loss of charging events. Also, certain auxiliary spray transients must be considered for portions of the charging system. See Reference 2 for details on the assumed temperatures and sequence of events of these transients.

Gas accumulation in this water system can result in water hammer, pump cavitation and pumping of non-condensable gas into the reactor vessel. These effects may result in the system being unable to perform its specified safety function. The NRC issued Generic Letter 2008-01, Managing Gas Accumulation, to address the issue of gas accumulation in this system. See UFSAR Section 1.8.5 for further information.

9.1.2 SYSTEM DESCRIPTION

9.1.2.1 General

The CVCS is shown in Figures 9-3 (Unit 1) and 9-24 (Unit 2). Coolant normally flows through the CVCS, as shown in Figures 9-3 and 9-24. Coolant letdown from one reactor coolant loop cold leg first passes through the tube side of a regenerative heat exchanger where the temperature is reduced to approximately 232°F and then through the letdown control valves. The letdown control valves, which are modulated by the pressurizer level control system, control the letdown flow to maintain proper pressurizer level. The letdown coolant temperature is reduced to 120°F at the letdown heat exchanger downstream of the letdown control valves. This temperature is selected to prevent deterioration of the ion exchange resins downstream. Flashing of the hot liquid between the letdown control valves and the letdown heat exchanger is prevented by controlling back pressure with a pressure control valve downstream of the letdown heat exchanger.

The cooled letdown next passes through one of two purification filters which remove suspended solids from the letdown before it enters an ion exchanger. The purified flow from the ion exchanger is sprayed into the Volume Control Tank (VCT) after passing through a strainer.

Charging pumps take suction from the VCT to add makeup coolant to the RCS via the shell side of the regenerative heat exchanger.

A small bypass flow around the purification filters passes through a process radiation monitor (to measure coolant activity). The proper flow rate is obtained by throttling the process radiation monitor outlet valve. There is also an indicating alarm on the discharge to monitor flow and alarm on a low flow condition.

If the level in the VCT reaches the high level setpoint, the letdown flow is automatically diverted to the liquid WPS. If the level in the VCT reaches the low-level setpoint, makeup water, borated to the existing concentration of the RCS, can be automatically supplied to the VCT. During normal operation when the level in the VCT reaches the low-low level setpoint, the tank discharge valve shuts and the suction of the charging pump is automatically aligned to the Refueling Water Tank (RWT).

With the level in the normal control band, the VCT has sufficient capacity to accommodate the variation in water inventory of the RCS due to power level changes in excess of that accommodated by the pressurizer.

Boric acid required for makeup can be supplied from either boric acid batching or from the boron recovery system. The boron recovery system is described in Section 11.1.2. The concentrated boric acid is stored in two heated storage tanks. Two pumps are provided to transfer concentrated boric acid. The piping is arranged such that the boric acid may be mixed with demineralized water in a predetermined ratio prior to being introduced to the VCT.

Chemicals are introduced to the RCS directly from a chemical addition tank or via a chemical addition metering pump, both of which are connected to the charging pump suction header.

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

9.1.2.2 Volume Control

The CVCS automatically adjusts the volume of water in the RCS using a signal from level instrumentation located on the pressurizer. The system reduces the amount of fluid that must be transferred between the coolant system and the CVCS during power changes by employing a programmed pressurizer level setpoint which varies with reactor power level. The setpoint varies linearly with the average reactor coolant temperature. This linear relationship is shown in Figure 4-10 (Section 4.3). The control system compares the programmed level setpoint with the measured pressurizer water level. The resulting error signal is used to control the operation of the charging pumps and the letdown valves, as described below. The pressurizer level control program is shown in Figure 4-11.

The pressurizer level control program regulates the letdown flow by adjusting the letdown control valve, so that the RCP 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 will start one or both standby charging pumps to restore level, and an increase in level will increase the letdown flow rate and initiate a backup signal to stop the two standby charging pumps.

The VCT coolant level can be automatically controlled. When the level in the tank reaches the high-level setpoint, the letdown flow is automatically diverted to the liquid waste processing system. If the makeup mode selector switch is in auto when the level in the tank reaches the low-level setpoint, makeup water is automatically supplied.

When the Control Room handswitches for the VCT outlet valve and RWT charging pump suction valve are in AUTO, and the level in the VCT reaches the low-low-level setpoint, the VCT outlet valve automatically closes and the RWT charging pump suction valve automatically opens, realigning the suction of the charging pumps to the RWT. On a loss of power to the level transmitter controlling this automatic action, the handswitch for the VCT outlet valve is placed in OPEN and the handswitch for the RWT charging pump suction valve is placed in CLOSE to reverse the automatic realignment that occurs. In this condition, VCT level can still be monitored in the Control Room from an indicator that is supplied from an independent power source.

The VCT can be vented to the WPS. The tank is normally operated with sufficient hydrogen partial pressure such that the RCS hydrogen concentration is consistent with plant chemistry requirements as discussed in Sections 4.1.4.2.3 and 9.1.2.3. However, other gases dissolved in the reactor coolant can leave solution when the letdown flow is sprayed into the VCT.

9.1.2.3 Chemical and Reactivity 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 resin bed which removes soluble impurities by ion exchange and suspended impurities by impaction of the particles on the surface of the resin beds.

Cartridge-type filters located upstream of the ion exchangers remove most of the suspended impurities to prevent clogging of the resin beds.

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

The reactor coolant is chemically conditioned to the typical conditions recommended in the EPRI PWR Primary Water Chemistry Guidelines:

a. Hydrazine scavenging to remove oxygen prior to exceeding 250°F;
b. Maintaining excess hydrogen concentration to control oxygen concentration and suppress radiolysis when the reactor is critical;
c. Chemical additives to control pH when the reactor is critical. As an exception to the Electric Power Research Institute Guidelines, the RCS lithium concentration may be as high as 5.33 ppm for approximately the first 4 effective full power days and 5.30 ppm for the remainder of the fuel cycle to optimize RCS pH.
d. Low levels of zinc acetate may be added to Units 1 and 2 for purposes of reducing dose and mitigating primary water stress corrosion cracking.
e. Hydrogen peroxide may be added at shutdown to promote dissolution of radiocobalt and scavenge hydrogen.

The chemical addition tank or chemical addition metering pump is used to feed chemicals to the charging pumps which inject the additives into the RCS. The reactor coolant makeup pumps can inject hydrazine into the makeup train to scavenge dissolved oxygen.

The CVCS is designed to prevent fission and corrosion product activities from exceeding the values given in Chapter 11 when operating with 1% failed fuel.

Reactivity Control

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

1. Optimize the position of the control rods;
2. Compensate for reactivity changes caused by variations in the temperature of the coolant, and by burnup of the core;
3. Provide a margin of shutdown for maintenance, refueling or emergencies.

The system includes a batching tank for preparing boric acid solution, two tanks for storing the solution, and two pumps for supplying boric acid solution to the makeup system. Boric acid from the waste processing system is pumped to either the boric acid storage or batching tanks.

Normally, the CVCS adjusts the boric acid concentration of the coolant by feed and bleed. To change concentration, the makeup (feed) system supplies either demineralized water or concentrated boric acid to the VCT or directly to the charging pump suction header, and the letdown (bleed) stream is diverted to the WPS. Toward the end of a core cycle, an ion exchanger is used to deborate. This avoids the excessive quantity of waste produced due to the feed and bleed operations.

The system can add boric acid to the reactor coolant 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 RCS at normal operating pressure when the plant is shut down. Leaks in the RCS may be detected while

the plant is at power by monitoring pressurizer level, VCT level, letdown flow, reactor coolant drain tank level, coolant temperature, and charging flow rate.

9.1.3 SYSTEM COMPONENTS

The major components of the CVCS and their functions are described in this section.

9.1.3.1 Description

Regenerative Heat Exchanger

The regenerative heat exchanger (Table 9-3) transfers heat from the letdown stream to the charging stream. Materials of construction are primarily austenitic stainless steel.

Letdown Control Valves

The letdown control valves (Table 9-4) regulate the reactor coolant flow from the regenerative heat exchanger as required by the pressurizer level regulating system. The valves reduce the pressure of the letdown fluid to about 460 psig. This value prevents flashing with about a 30 psi margin, even with minimum makeup flow (44 gpm charging) and maximum letdown flow (128 gpm) [Table 9-1, note [a]]. The letdown flow is nominally 38 gpm, for coolant purification, but will vary as the pressurizer water level changes. The valves are pneumatically-operated and fail closed. All parts in contact with reactor coolant are of austenitic stainless steel.

Letdown Heat Exchanger

The letdown heat exchanger (Table 9-5) cools the letdown stream in the tube side of the regenerative heat exchanger to a temperature suitable for entry into a purification ion exchanger. Component cooling system fluid is the cooling medium on the shell side of the letdown heat exchanger. Tube side materials of construction are primarily austenitic stainless steel; shell side materials of construction are primarily carbon steel [Table 9-1, note [a]].

Ion Exchangers

Three purification ion exchangers (Table 9-6) are available to purify and remove boron from the reactor coolant. These ion exchangers are identical in design and may be interchanged during operation. Each unit is designed to handle the maximum letdown flow of 128 gpm [Table 9-1, note [a]]. The vessels and resin retention element are of austenitic stainless steel construction.

Mixed bed resin is loaded into an Ion Exchanger to purify the reactor coolant by removing corrosion and fission products. Toward the end of core life, resin is loaded into one or more ion exchangers to reduce boron concentration in the reactor coolant. This method is preferable to using feed and bleed since it minimizes the volume of radioactive waste water produced.

Purification Filters

The purification filters (Table 9-7) remove suspended impurities from the reactor coolant. Each filter will accommodate maximum letdown flow of 128 gpm. The filter housings are austenitic stainless steel.

Volume Control Tank (VCT)

The VCT (Table 9-8) accumulates water from the RCS. The tank has sufficient capacity to accommodate the variation in water inventory of the RCS due to power

level changes in excess of that accommodated by the pressurizer. The tank provides a gas space where a partial pressure of hydrogen and nitrogen is maintained to control the hydrogen and nitrogen concentration in the reactor coolant. A vent to the WPS permits removal of hydrogen, nitrogen and gaseous fission products released from solution in the VCT. The tank is of austenitic stainless steel construction and provided with overpressure protection. Level controls divert coolant to the WPS on high level or operate coolant makeup valves on low level.

With respect to quality control (QC) the CVCS volume control tanks, purchased using Combustion Engineering, Inc. (CEs) generic specification WQC-11.1, Level II, required the following:

a. The manufacturer was required to maintain a quality assurance system acceptable to CE. This system included inspection and testing procedures, a manufacturing and QC plant, control of procedure revisions and control and submittal of documents and records.
b. The manufacturer was required to have written procedures which ensured the latest applicable drawings, specifications and instructions were used for fabrication, inspections and tests and ensured control over all measuring and testing equipment.
c. The manufacturer was responsible for assuring that all supplies and services procured from his suppliers (sub-contractors and vendors) conformed to the contract requirements.
d. The QC program of the manufacturer was required to ensure that raw material to be used in fabrication or processing products conformed to the applicable physical, chemical and all other technical requirements. The identification of all material was maintained throughout all operations by job number, lot number, heat number or any other suitable identification means and recorded on proper inspection records for each component.
e. The manufacturer was informed that all processing, testing and insertion operations taking place in the supplier's or subcontractor's facilities were subject to CE/Baltimore Gas and Electric Company (BGE) quality surveillance and verification.

<u>Charging Pumps</u>

Three positive displacement charging pumps (Table 9-9) supply makeup water to the RCS. The pumps return coolant to the RCS. On a SIAS, all three pumps are started and discharge concentrated boric acid into the RCS. All wetted parts, except seals, are of stainless steel and titanium. The charging pumps have a design flow of 44 gpm each.

<u>Concentrated Boric Acid Tanks</u>

Each of the two concentrated boric acid tanks (Table 9-10) stores enough concentrated boric acid solution to bring the reactor to a cold shutdown condition at any time during the core lifetime. The solution is either prepared in the boric acid batching tank and flows through the boric acid batching strainer before entering the storage tanks or is obtained from the boron recovery system. The combined capacity of the tanks will also be sufficient to bring the coolant to refueling concentration. The tanks have duplicate electric heaters to maintain a temperature above the saturation temperature of the concentrated solution, and sampling connections are used to verify that proper concentration is maintained. The tanks are constructed of stainless steel.

Boric Acid Pumps

The two boric acid pumps (Table 9-11) supply concentrated boric acid solution through the boric acid strainer (Table 9-11) to the makeup system where the boric acid may be diluted with demineralized water. On receipt of SIAS, these pumps line up with the charging pumps to permit direct introduction of concentrated boric acid into the RCS. Each is capable of supplying boric acid at the maximum demand conditions. Wetted parts of the pumps are stainless steel.

Process Radiation Monitor

The process radiation monitor (Table 9-13) continuously measures the activity of the reactor coolant and actuates an alarm in the Control Room if a predetermined activity level is reached. The sensor is a gross-gamma plus specific isotope (I-135) monitor; the system is designed to detect activity release from the fuel to the reactor coolant within five minutes of the event.

9.1.3.2 Codes and Standards

All components are designed, manufactured, tested and inspected according to applicable codes. The following code classifications apply to the CVCS components:

Regenerative Heat Exchanger	ASME III Class C[a]
Letdown Heat Exchanger	ASME III Class C
Deborating Purification Demineralizers	ASME III Class C
Purification Filters	ASME III Class C
Volume Control Tank	ASME III Class C
Boric Acid Storage Tanks	ASME III Class C

[a] The regenerative heat exchanger is built as a Class A vessel, but is stamped Class C.

9.1.3.3 Testing and Inspection

Each component is inspected and cleaned prior to installation into the system. Demineralized water will be used to flush each system.

Instruments will be calibrated during testing. Automatic controls will be tested for actuation at the proper setpoints. Alarm functions will be checked for operability and limits during preoperational testing. The relief valve setpoints will be checked.

The system will be operated and tested initially with regard to flow paths, flow capacity and mechanical operability. At least one pump of each type will be tested to demonstrate head and capacity.

Data will be taken periodically during normal plant operation to confirm heat transfer capabilities and purification efficiency.

9.1.4 SYSTEM OPERATION

9.1.4.1 Startup

During startup, the plant is brought from cold shutdown to hot standby at normal operating pressure and zero power temperature, before the reactor is brought critical. While the coolant is being heated, and until the pressurizer steam bubble is established, the charging pumps and letdown backpressure valve are used to maintain pressure in the RCS. After a steam bubble is established in the

pressurizer, the operator adjusts the pressurizer water level manually with the letdown backpressure and letdown control valves. The level controls of the VCT automatically divert the letdown flow to the WPS.

While the reactor is shut down, the VCT can be vented to the WPS. Prior to startup, the tank is purged with nitrogen to remove air. After purging is completed, the vent is secured and a nitrogen-hydrogen blanket is established in the tank. Any oxygen in the reactor coolant is normally removed by radiolytic recombination with excess hydrogen in the coolant. However, should the residual radiation from the core be insufficient to reduce the oxygen level, hydrazine can be added to scavenge the oxygen if the temperature is below 250°F.

Throughout startup, one purification filter is in service to reduce the activity of wastes entering the WPS. When the letdown temperature is stabilized at the desired RCS hot standby temperature, one or more purification ion exchangers are put into service as required.

Within limitations placed on the 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 makeup controller "Dilute" mode. The concentration of boric acid in the reactor coolant is determined by chemical analysis.

For the initial reactor startup following refueling, the RCS soluble boron concentration shall remain at or above refueling boron concentration until all four RCPs are running with RCS $T_{avg} \geq 515°F$ in Mode 3. After meeting those conditions, the RCS boron concentration may be reduced to that required to satisfy Mode 3 Shutdown Margin with no credit taken for the highest CEA bank worth. After successfully completing CEA rod drop time testing and with all shutdown CEA banks in the fully-withdrawn position, then additional RCS dilution may proceed.

9.1.4.2 Normal Operation

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

During normal operation:

a. Level instrumentation on the pressurizer automatically controls the volume of water in the reactor system by adjusting the letdown flow.
b. The VCT level is increased manually by the operator using makeup and automatically decreased by diversion to the WPS. Level can also be controlled by automatic makeup.
c. The hydrogen concentration and pH of the coolant are adjusted.
d. 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 WPS. Late in cycle life, the dissolved boron in the reactor coolant is maintained at a very low concentration; at this time, feed-and-bleed generates excessive radioactive wastes; further reduction is accomplished by use of a purification ion exchanger with deborating resin. 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 VCT or directly to the charging pump

suction header 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" mode, the flows of the demineralized water and concentrated boric acid are set for any blend concentration between demineralized makeup water and concentrated boric acid. This mode is primarily used to supply the VCT. It is also used for positive reactivity control during power operation.

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 VCT according to signals received from the VCT level program.

e. 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.

9.1.4.3 Cooldown

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

Before the plant is cooled down, the VCT is vented to the WPS to reduce the activity and the hydrogen concentration in the reactor coolant. The operator may also increase the letdown flow rate to accelerate degasification, ion exchange and filtration of the reactor coolant. The operator increases the concentration of boric acid in the reactor coolant to ensure that the reactor has an adequate shutdown margin throughout its period of cooldown.

During cooldown, makeup water is introduced at the shutdown boric acid concentration. When the CVCS makeup system is in the automatic mode, a preset boric acid solution is automatically blended and introduced into the VCT upon demand from the VCT level program. The preset solution concentration corresponding to the desired shutdown concentration will have been previously determined and selected on the blender switch by the operator. During the cooldown, the charging pumps and letdown control valves are used to adjust and maintain the pressurizer water level. High charging flow results in a low level in the VCT which sounds an alarm. The operator then manually makes up fluid volume at the preselected shutdown boric acid concentration.

The estimated dissolved boron in the reactor coolant required to maintain cold shutdown conditions is shown in curves found in the Nuclear Engineering Operating Procedure (NEOP).

The total volume of both concentrated boric acid storage tanks is also sufficient to bring the RCS to refueling concentration.

A portion of the charging flow is used as an auxiliary spray to cool the pressurizer when the pressure of the RCS is below that required to operate the RCPs.

9.1.4.4 Safety Injection

Under event conditions, the charging pumps are used to inject concentrated boric acid into the RCS. Either the pressurizer level control system or SIAS will automatically start all charging pumps. The SIAS will also function to transfer the charging pump suction from the VCT 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 tanks to the charging pump suction header. If the charging line inside the reactor containment building is inoperative, the line may be isolated outside the reactor containment, and the concentrated boric acid solution may be injected by the charging pumps through the high-pressure safety injection (HPSI) piping.

9.1.5 DESIGN EVALUATION

To assure reliability, the design of the CVCS incorporates redundant critical components to reduce dependence upon any single critical component. Redundancy is provided as follows:

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

The charging and boric acid pumps are powered by the diesel generators if normal power sources are lost. One charging pump and one boric acid pump are supplied from each emergency bus. The third charging pump may be supplied from either emergency bus. 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 is running after SIAS. If two diesel generators are available, both boric acid pumps will be running. The charging pumps and boric acid pumps may be controlled locally at their switchgear. Separate power supplies for pump power and separate control circuits assure that this system satisfies the single failure criterion.

The boric acid solution is stored in heated and insulated tanks and is piped in heat-traced and insulated lines to preclude precipitation of the boric acid. Two independent and redundant heating systems are provided for the boric acid tanks and lines. Low temperature alarms and automatic temperature controls are included in the heating system. 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 SI system to inject concentrated boric acid into the RCS.

9.1.6 REFERENCES

1. Bechtel Specification 6750-M-0310C, "Design Specification for Piping, Valves, and Associated Equipment of the Letdown System"
2. Bechtel Specification 6750-M-0310D, "Design Specification for Piping, Valves, and Associated Equipment of the Charging System and Auxiliary Spray System"

TABLE 9-1

CHEMICAL AND VOLUME CONTROL SYSTEM PARAMETERS

Normal Letdown and Purification Flow, gpm[a]	38
Normal Charging Flow, gpm	44
Reactor Coolant Pump Controlled Bleedoff (4 pumps), gpm[a]	6
Normal Letdown Temperature at Loop °F	548
Ion Exchanger Operating Temperature, °F	120

[a] The original design for normal letdown and purification flow rate was 40 gpm with 4 gpm of reactor coolant pump controlled bleedoff. (The maximum letdown and purification flow rate was 128 gpm with 4 gpm controlled bleedoff.) These flows combined to equal the normal charging flow rate of 44 gpm (maximum charging flow rate of 132 gpm). Facility Change Request (FCR) 87-0074 replaced the reactor coolant pump seal with a state-of-the-art design that requires 1.5 gpm (nominal) controlled bleedoff per pump for stable operation. The net effect of this FCR was to reduce the normal letdown and purification flow rate to 38 gpm (the maximum letdown and purification flow rate was reduced to 126 gpm) and increase the reactor coolant pump controlled bleedoff rate to 6 gpm (total for all four pumps).

TABLE 9-3

REGENERATIVE HEAT EXCHANGER

DESIGN PARAMETERS

Quantity	1
Type	Shell and Tube, Vertical
Code	ASME III, Class C[(a)]
Tube Side (Letdown) Fluid	Reactor Coolant, 1.5 wt% Boric Acid, Maximum
Design Pressure, psig	2485
Design Temperature, °F	650
Materials	Stainless Steel, Type 304
Pressure Loss at 63,500 lb/hr	99
Shell Side (Charging) Fluid	Reactor Coolant, 6.25 wt% Boric Acid, Maximum
Design Pressure, psia	3025
Design Temperature, °F	650
Materials	Stainless Steel, Type 304
Pressure Loss at 132 gpm, psi	70

(a) The regenerative heat exchanger is built as a Class A vessel, but is stamped Class C.

DESIGN OPERATING PARAMETERS - REGENERATIVE HEAT EXCHANGER

TUBE SIDE (LETDOWN)	NORMAL	MAXIMUM UNBALANCED CHARGING WITH HEAT TRANSFER	MAXIMUM PURIFICATION	MAXIMUM UNBALANCED LETDOWN
Flow – gpm [Table 9-1 note (a)]	38	30	126	126
Inlet Temp. - °F	548	548	548	548
Outlet Temp. - °F	232	143	350	433
Shell Side (Charging)				
Flow – gpm	44	132	132	44
Inlet Temp. - °F	120	120	120	120
Outlet Temp. - °F	415	220	324	475

TABLE 9-4

LETDOWN CONTROL VALVES

Quantity	2
Design Pressure, psia	2500
Design Temperature, °F	650
Flow, each	
Maximum, gpm	128
Minimum, gpm	29

TABLE 9-5

LETDOWN HEAT EXCHANGER

DESIGN PARAMETER

Quantity	1
Type	Shell and Tube, Horizontal
Code	ASME III, Class C
Tube Side (Letdown) Fluid	Reactor Coolant, 1.5 wt% Boric Acid, Maximum
Design Pressure, psig	650
Design Temperature, °F	550
Pressure Loss at 63,500 lb/hr, psi	52
Materials	Stainless Steel, Type 304
Shell Side (Cooling Water)	
Fluid	Component Cooling Water
Design Pressure, psig	150
Design Temperature, °F	250
Materials	Carbon Steel
Design Flow, lb/hr	594,390

DESIGN OPERATING PARAMETERS

TUBE SIDE (LETDOWN)	NORMAL	MAXIMUM UNBALANCED CHARGING WITH LETDOWN	MAXIMUM PURIFICATION	MAXIMUM UNBALANCED LETDOWN
Flow - gpm [Table 9-1 note (a)]	38	30	126	126
Inlet Temp. - °F	232	143	350	433
Outlet Temp. - °F	120	120	125	135
Shell Side (Cooling Water)				
Flow - gpm	157	21	1200[a]	1200[a]
Inlet Temp. - °F	95	65	95	95
Outlet Temp. - °F	122	128	118	127

[a] This flowrate represents design points selected during the design of the system and components. It does not indicate the normal operating point or minimum or maximum limitation of the system or components.

TABLE 9-6

ION EXCHANGERS

Quantity	3
Type	Flushable
Design Pressure, psig	200
Design Temperature, °F	250
Normal Operating Pressure, psig	60
Normal Operating Temperature, °F	120
Resin Volume, ft^3, each	36
Normal Flow, gpm	38
Maximum Flow, gpm	128 [Table 9-1, note (a)]
Code for Vessel	ASME III, Class C
Material	ASME SA 240, Type 304
Fluid, wt% Boric Acid, Maximum	1.5

TABLE 9-7

PURIFICATION FILTERS

Quantity	2
Type of Elements	Single Element Disposable Cartridge
Filter Rating, microns (absolute)	0.1 to 6.0, various, depending on plant conditions
Vessel Design Pressure, psig	200
Vessel Design Temperature, °F	250
Design Flow, gpm	128 (Table 9-1, note (a)]
Normal Flow, gpm	38
Code for Vessel	ASME III, Class C
Material	Austenitic Stainless Steel
Fluid, wt% Boric Acid, Maximum	1.5

TABLE 9-8
VOLUME CONTROL TANK

Quantity	1
Type	Vertical, Cylindrical
Design Pressure, Internal, psig	75
Design Pressure, External, psig	15
Design Temperature, °F	250
Operating Pressure Range, psig	0 to 65
Normal Operating Pressure, psig	25 to 50
Normal Operating Temperature, °F	120
Normal Spray Flow, gpm	38
Blanket Gas	Hydrogen and/or Nitrogen
Code	ASME III, Class C
Fluid, wt% Boric Acid, Maximum	12
Material	Austenitic Stainless Steel

TABLE 9-9

CHARGING PUMPS

Quantity	3
Type	Positive Displacement
Design Pressure, psig	2735
Design Temperature, °F	250
Capacity, gpm	44
Normal Discharge Pressure, psig	2311
Normal Suction Pressure, psig	50
Normal Temperature of Pumped Fluid, °F	120
Maximum Discharge Pressure (Short Term), psig	3010
Minimum NPSH, psia	9
Driver Rating, hp	100
Materials in Contact with Pumped Fluid	Stainless Steel, Titanium
Fluid, wt% Boric Acid, Maximum	12

TABLE 9-10

CONCENTRATED BORIC ACID PREPARATION AND STORAGE

CONCENTRATED BORIC ACID TANKS

Quantity	2
Internal Volume, ft^3	1270
Design Pressure, psig	15
Design Temperature, °F	200
Normal Operating Temperature, °F	150
Type Heater	Duplicate Electrical, Strap-on heaters
Fluid, wt% Boric Acid, Maximum	12
Material	Stainless Steel
Code	ASME III, Class C

BORIC ACID BATCHING STRAINER

Quantity	1
Type	Basket
Design Pressure, psig	150
Design Temperature, °F	200
Screen Size, US Mesh	80
Design Flow, gpm	50
Materials	Stainless Steel
Fluid, wt% Boric Acid, Maximum	12

BORIC ACID BATCHING TANK

Quantity	1
Useful Volume, ft^3	67
Design Pressure	Atmospheric
Design Temperature, °F	200
Normal Operating Temperature, °F	150
Type Heater	Electrical Immersion
Heater Capacity, min, kW	45
Fluid, wt% Boric Acid, Maximum	12
Material	Austenitic Stainless Steel

TABLE 9-11

BORIC ACID PUMPS AND STRAINER

PUMPS

Quantity	2
Type	Centrifugal
Design Pressure, psig	150
Design Temperature, °F	250
Design Head, ft	231
Design Flow, gpm	143
Normal Operating Temperature, °F	150
NPSH Required, ft	20
Horsepower	25
Fluid, wt% Boric Acid, Maximum	12
Material in Contact With Liquid	Stainless Steel

STRAINER

Quantity	1
Type	Basket
Screen Size US Mh	80
Design Pressure, psig	150
Design Temperature, °F	200
Design Flow, gpm	140
Materials	Austenitic Stainless Steel
Liquid, wt% Boric Acid, Maximum	12

TABLE 9-13

PROCESS RADIATION MONITOR

Quantity	1
Design Pressure, psig	200
Design Temperature, °F	250
Normal Operating Pressure, psig	80
Normal Operating Temperature, °F	120
Normal Flow Rate, gpm	0.5
Measurement Range, μCi/cc I-135	10^{-4} to 100[a]
Measurement Range, cpm	10 to 10^6

[a] Upper measurement range of 100 μCi/cc is based upon use of a collimator between detector and sample.

9.2 SHUTDOWN COOLING SYSTEM

9.2.1 DESIGN BASIS

The Shutdown Cooling (SDC) System is used to remove core decay heat and reactor coolant sensible heat during plant cooldowns and cold shutdowns. The system also cools the containment spray water during Containment Spray System (CSS) operation following a Recirculation Actuation Signal (RAS) and maintains RCS temperature during refueling operations. Additionally, the heat exchangers can be used to provide additional spent fuel pool cooling (SFPC) when the complete core is removed from the reactor vessel and temporarily stored in the spent fuel pool (SFP).

Portions of the SDC System piping are ASME Class 1, and thus require a fatigue analysis of the applicable thermal shock transients, and other operational cycles. In addition to design cyclic transients a, d, and e of Section 4.1.1, the SDC System fatigue analysis considers 500 initiations of SDC. In this transient, 300°F water from the RCS is injected into the SDC piping which is at a higher temperature due to the residual effects of the RCS normal operating temperatures. See Reference 1 for further details.

Gas accumulation in this water system can result in water hammer, pump cavitation and pumping of non-condensable gas into the reactor vessel. These effects may result in the system being unable to perform its specified safety function. The NRC issued Generic Letter 2008-01, Managing Gas Accumulation, to address the issue of gas accumulation in this system. See UFSAR Section 1.8.5 for further information.

9.2.2 SYSTEM DESCRIPTION

The SDC system is shown schematically in Figure 9-5. The system uses portions of other systems, i.e., the RCS (Section 4.1) and the engineered safeguards (Sections 6.3 and 6.4).

In the SDC mode of operation, reactor coolant is circulated through the tube side of the SDC heat exchanger using the low-pressure safety injection (LPSI) pumps. The flow path from the pump discharge runs through normally locked-closed valve, SI 658, through the shutdown cooling heat exchangers, and through normally locked-closed valve, SI 657, to the LPSI header, and enters the RCS through the four safety injection nozzles. The circulating fluid flows through the core and is returned from the RCS through the SDC nozzle in the loop No. 2 reactor vessel outlet (hot leg) pipe. The coolant is returned to the suction of the LPSI pumps through normally locked-closed valves SI-651 and SI-652.

In Mode 5, 6, or defueled, a containment spray pump may be used to circulate reactor coolant through the tube side of the SDC heat exchangers. The appropriate valve lineup and plant operating conditions are specified in the Operating Procedures.

During CSS operation, prior to recirculation, RWT inventory passes through the tube side of the SDC heat exchanger via containment spray pumps for containment cooling purposes. After recirculation, the containment spray pumps switch suction from the RWT to the containment sump and sump water is circulated through the SDC heat exchangers.

In both the SDC mode of operation and during CSS operation, component cooling (CC) water flows through the shell side of the SDC heat exchangers. During shutdown cooling, CC cools reactor coolant and during containment spray operation after RAS, CC cools the containment sump fluid. Prior to RAS, CC is not needed for cooling purposes because RWT water does not need cooling. Also note that prior to RAS, CC is not cooled by Saltwater, so CC could provide no cooling.

Shutdown cooling and total low-pressure injection flow are measured by a flow element installed in the low-pressure injection header. Flow is indicated in the Control Room. The flow element also transmits a signal to a controller which will provide automatic flow control during SDC operation.

9.2.3 SYSTEM COMPONENTS

The SDC system is made up of portions of the SI system and the RCS. The principal characteristics of the major components in those systems are given in Sections 6.3 and 4.1, respectively.

Each component is inspected and cleaned prior to installation. Demineralized water is used to flush each system. Initially the system is operated and tested to verify that the flow path, flow, thermal capacity and mechanical operability meet the design requirements. Instruments are calibrated during testing. The automatic flow control is tested.

Periodic testing of the LPSI pumps, as described in Section 6.3.6, assures the availability of this equipment for shutdown cooling. Data can be taken during refueling operations to confirm heat transfer capacity.

9.2.4 SYSTEM OPERATION

During normal plant operation, there are no components of the system in operation. All components are on standby for possible emergency operation, as a part of the CSS and SI system. The SDC capability may be used during the early stages of plant startup to control the reactor coolant temperature. As the coolant temperature approaches 300°F and the pressure approaches 270 psig, this method of control must be discontinued and the system aligned for emergency operation.

Following reactor shutdown and cooldown, the system is operated in the shutdown mode for further cooling of the RCS when the coolant temperature falls below 300°F and the coolant pressure falls below 270 psig. At this time, the system must be manually realigned for shutdown cooling. Prior to placing the system in operation, the boron concentration is verified at various points in the system. During the early stages of shutdown cooling, the cooldown rate is controlled by limiting the flow through the tube side of the heat exchanger. Constant flow through the core is maintained by using valve SI 306 as a heat exchanger bypass valve.

During Mode 6 operation, the SDC system serves to remove decay heat and other residual heat from the RCS, provide mixing of borated coolant, to provide sufficient coolant circulation to minimize the effects of a boron dilution accident, and to prevent boron stratification. Heat is removed from the RCS by circulating reactor coolant through the SDC heat exchangers where its heat is transferred to the component cooling water system which is, in turn, cooled by the Saltwater System. Therefore, in Mode 6, an OPERABLE SDC loop requires the support of a functional component cooling and saltwater subsystem. In Mode 6 conditions where both SDC loops are required to be operable (water level < 23 feet above the tops of the irradiated fuel assemblies seated in the reactor vessel), only one functional component cooling and one saltwater subsystem is necessary, provided their heat removal capacity is sufficient.

9.2.5 SYSTEM PERFORMANCE

The SDC system is designed to reduce the temperature of the reactor coolant at the controlled cooldown rate from 300°F to refueling temperature (≤ 140°F) within 36 hours

after shutdown. This assumes both CC pumps, both CC heat exchangers, and both SDC heat exchangers are on line and CC reaches a maximum of 120°F. This further assumes that each LPSI is circulating 3000 gpm and each CC pump is circulating 2500 gpm through the SDC heat exchanger (Section 6.3.2.5). Cooldown will occur more or less rapidly depending on pump and heat exchanger availability and component cooling loads on line.

The SDC system is designed to cool containment spray flow in order to bring containment temperature down to 120°F within 30 days following an accident. This assumes minimum safeguards: one train of containment spray, one train of safety injection, and one train of containment air coolers. This also assumes a CC flow of 1800 gpm and a containment spray flow of 1250 gpm.

9.2.6 DESIGN EVALUATION, AVAILABILITY, AND RELIABILITY

During normal cooldown the system utilizes the LPSI pumps to circulate the reactor coolant through the two SDC heat exchangers, returning it to the RCS through the LPSI header. Cooldown rate is controlled by adjusting the flow through the heat exchangers. Both heat exchangers are required to achieve cooldown at the maximum design rate. One exchanger provides cooldown capability at a reduced rate.

Control valves which were originally equipped with two sets of packing and intermediate leakoff connections that discharged to the WPS were repacked with Chesterton packing. Valves that are repacked with Chesterton packing have one set of packing. These valves may have their leakoff lines removed and the valve leakoff connection plugged or tubing capped. Some manual valves have backseats to facilitate repacking.

All piping in the SDC system is austenitic stainless steel. The piping is welded except for flanged connections at the pumps and components, which can be removed for maintenance.

During plant operation, double valves with a relief valve between the two valves, isolate the suction of both LPSI pumps from the RCS. These two valves, SI-651 and 652, are key-locked closed at the control board during plant operation. Additionally, the valve between the SDC heat exchangers and the LPSI header, SI-657, is locked shut during plant operation, both locally and at the control board. The keys are kept under administrative control to ensure that these valves cannot be opened inadvertently during plant operation.

Pressurizer pressure instrument channels P-103 and P-103-1 each provide an open permissive interlock to the two LPSI pump suction isolation valves SI-651 and 652, respectively. These independent and redundant interlocks prevent opening of these valves whenever the RCS is already pressurized at or above the SDC System design pressure. The suction piping to the LPSI pumps is the SDC System component with the limiting design pressure rating.

During SDC System operation, a visual and audible alarm on the main control board is activated whenever either SI-651 or 652 are not fully closed and RCS pressure is above the SDC System design pressure. These two separate alarms are tested at each refueling outage to ensure reliability and are designed to alert the operator in the event of an alarm or control circuit power supply failure. These alarms, associated procedural controls and operator training ensures a high probability of achieving double isolation of the SDC System from the RCS when the RCS pressure is raised above the SDC System design pressure.

The suction isolation valves, SI-651 and SI-652, and associated control system design, therefore, provide two independent and redundant means for achieving and maintaining isolation of the SDC System from the RCS.

Overpressure protection of the SDC System is provided by relief valve RV-468, which is located on the SDC Return Header downstream of 1(2)-MOV-651. This valve is sized to protect the SDC flowpath from overpressure due to the simultaneous operation of three charging pumps while on SDC with the pressurizer in a solid condition.

Certain transients in the RCS, such as an inadvertent RCP or HPSI pump start, can cause a pressure transient that exceeds the capacity of RV-468. However, these transients are prevented or mitigated by the LTOP controls outlined in Section 4.2.2 and in the Technical Specifications. The LTOP controls are in place at all times when both SDC is in operation and a pressurization event is possible (e.g., until the RCS is vented to at least 8 square inches).

9.2.7 OPERATION AT REDUCED INVENTORY

Generic Letter 88-17, Loss of Decay Heat Removal, described concerns and recommended actions for operation of the RCS and the SDC System during reduced inventory conditions. Reduced inventory is defined by the generic letter as an RCS inventory which results in a reactor vessel level lower than 3’ below the vessel head flange. Three key areas were addressed in the resolution of this issue: (1) Prevention of a loss of decay heat removal; (2) In-depth mitigation of a loss of decay heat removal; and (3) Providing a closed containment before the core uncovers if a loss of decay heat removal occurs. These three areas were addressed in responses to the recommendations made in the generic letter. The recommendations and their responses are provided below:

a. Provide reliable indication of parameters that describe the state of the RCS and the performance of systems normally used to cool the RCS for both normal and accident conditions. At a minimum, provide the following in the Control Room:

   1. Two independent RCS level indications.

      We have at least two independent, continuous level indications and audible alarms. (Section 7.5.9.4)

   2. At least two independent temperature measurements representative of the core exit whenever the reactor vessel head is located on top of the reactor vessel.

      We have two independent, continuous coolant temperature indications that are representative of the core exit conditions. (Section 7.5.9.4)

   3. The capability of continuously monitoring decay heat removal system performance whenever the system is used for cooling the RCS.

      We have instrumentation to monitor SDC pump suction pressure, discharge pressure, motor current, system flow and RCS level. (Section 7.5.9.4)

   4. Visible and audible indications of abnormal conditions in temperature, level and decay heat removal system performance.

      We have visible and audible indications in the Control Room for temperature, level and SDC performance. (Section 7.5.9.4)

b. Develop and implement procedures that cover reduced inventory operation and that provide an adequate basis for entry into a reduced inventory condition. Procedures should cover normal and off-normal operation of the Nuclear Steam

Supply System during times when cooling is normally provided by the SDC System.

There are procedures and administrative controls which cover normal and off-normal operation of the RCS, SDC, supporting systems and containment. Additionally, we have implemented controls that ensure the status of each containment penetration required for containment closure is known, and the time and method of closure has been addressed for those penetrations which are open. The definition of containment closure is consistent with Technical Specifications.

c. Ensure adequate equipment is operating, operable and/or available to provide cooling for the RCS. Adequate equipment must remain operable or available to mitigate a loss of SDC. In addition, adequate equipment must be provided for personnel communications for activities necessary to maintain the RCS in a stable and controlled condition.

   Normally, the SDC System provides cooling for the RCS. Technical Specifications require that the SDC System remain operable. The HPSI pump and one other means remain available to mitigate the consequences of a loss of SDC. The normal plant paging system provides communication capability onsite. Any page can be overridden by a Control Room page which is simultaneously broadcast to all zones.

d. Conduct analyses to supplement existing information and develop a basis for procedures, instrument and installation and response, and equipment/Nuclear Steam Supply System interactions and response. The analysis should encompass thermodynamic and physical conditions, and should emphasize complete understanding of Nuclear Steam Supply System behavior.

   Analyses have been performed and include: time to reach saturated conditions; peak pressurization of the RCS based on reactor vessel head vent paths; times to reach core uncovery for a variety of conditions, assuming no operator action; effects of steam generator nozzle dam installation; instrument uncertainties; and analyses of flow paths as a function of RCS back pressure. Containment response and airborne activity analyses were also performed. All calculations are dependent upon initial conditions, such as: RCS heat sinks; level; temperature; vent paths; and time after shutdown.

e. The Technical Specifications should not restrict the safety benefit of actions identified under Generic Letter 88-17.

   We reviewed our Technical Specifications and made the necessary changes.

   Subsequent to the original Generic Letter 88-17 responses, a containment outage door was installed at the exterior of the equipment hatch opening as an additional programmed enhancement to provide for closure during Mode 5 and 6 conditions. The containment outage door is designed to mitigate the offsite radiological consequences of a fuel handling incident and a loss of shutdown cooling incident. Because it can be opened and shut more quickly than the equipment hatch, the containment outage door increases the availability of the equipment hatch opening for access to the Containment during unit outages.

9.2.8 REFERENCES

1. Bechtel Specification 6750-M-0310A, "Design Specification for Piping, Valves, and Associated Equipment of the Shutdown Cooling System"

9.3 CIRCULATING WATER SYSTEM

9.3.1 DESIGN BASIS

The condensers for both of the electrical generating units have been designed such that the increase in temperature of the Chesapeake Bay water passing through them is not more than 10°F at maximum expected, not guaranteed, operating conditions. Under guaranteed operating conditions, i.e., that maximum operating condition at which both the reactor supplier and the turbine generator suppliers guarantee their equipment, the temperature increase of the Bay water is no more than 9.6°F. A test program allowing a temperature increase up to 12°F has been completed by the State of Maryland. Current limits allow use of a 12°F increase on a permanent basis.

Circulating water pumps and piping conduits are designed to fulfill the design basis requirements described above.

9.3.2 SYSTEM DESCRIPTION

The circulating water system (CWS) is shown in Figures 9-8 (Unit 1) and 9-26 (Unit 2). The intake and discharge are shown on Figures 1-3A and 1-3B.

9.3.2.1 Circulating Water System

The CWS incorporates design information developed from the model testing discussed in Section 2.8.2. The full width of the intake channel (which serves both units) is 560'. The Intake Structure houses a total of 24 circulating water screens, 12 for each unit, consisting of single-flow or dual-flow designs. The purpose of these screens is to prevent debris larger than 3/8" or 10 mm from passing into the circulating water pumps, condenser, saltwater pumps, and the saltwater-to-fresh-water heat exchangers. The Screen Wash Systems provide a high pressure spray to remove debris from the water screens. The screen wash systems consist of eight submersible screen wash pumps including two installed spares and two submersible trough wash pumps. The screen wash pumps serve four screens each and the trough wash pumps serve the trough for each unit.

The Circulating Water Chemical Addition System serves both the Unit 1 and Unit 2 Circulating Water Systems to minimize the marine fouling of piping and heat exchanger surfaces. This system has the ability to inject approved chemicals into each of the Circulating Water System intake and discharge conduits, as necessary.

The CWS has six vertical centrifugal pumps per unit. These pumps provide the motive force required to circulate bay water through the system and back into the bay.

9.3.2.2 Condensers

The condensers for each unit consist of three separate shells, each with the same capacity, to condense exhaust steam from the power generating turbine. The condensers are of the single-pass or once-through design with divided water boxes to permit one-half of each shell to be opened and manually cleaned during plant operation, if necessary.

Each condenser has approximately 49,500 tubes, each being 1-1/4" in diameter and 28' long. The tube material is austenitic stainless steel and titanium (Unit 1), and titanium (Unit 2).

Each condenser is equipped with a mechanical cleaning system utilizing small sponge rubber balls which are injected at the condenser inlet, passed through the tubes, collected at the condenser outlet, and returned for recycling.

A butterfly valve equipped with a perforated disc instead of a solid disc is installed in each circulating water pipe at the inlet to the condenser water boxes. It is possible to close this valve when its corresponding circulating water pump is shut down. Marine growth that may have been pumped against the condenser tube sheet should fall off and be caught by this strainer-type valve instead of falling further down and out of reach in the pipes. Conveniently located manhole doors can then be opened and the marine growth manually removed from the condenser.

Two temperature sensors are provided in each discharge pipe. These temperatures are monitored by the computer in the Control Room. Since each unit has six discharge pipes, circulating water discharge is monitored by twelve independent temperature readings prior to being discharged into the bay.

9.3.2.3 Bay Water Systems Discharges

At mean low tide level, the top of the discharge conduits is approximately 6 feet below the surface of the water. The entire discharge structure is composed of four separate conduits, two for each unit.

The effluent from the WPS may be discharged into any one or more of these four separate conduits, thus providing a means to ensure a maximum dilution of the effluent under all operating conditions. Discharge from the condensate system, the steam generator blowdown recovery system, the storm water system, the yard oil interceptor, and the auxiliary blowdown tank are also directed to the discharge conduits.

9.3.3 COMPONENTS

The component description for the CWS is contained in Table 9-15.

9.3.4 TESTING AND INSPECTION

Each component is inspected and cleaned prior to installation into the system.

Instruments are calibrated during testing. Automatic controls are tested for actuation at the proper setpoints. Alarm functions are checked for operability and limits during preoperational testing. The relief valve setpoints are checked.

The system was operated and tested initially with regard to flow paths, flow capacity and mechanical operability.

Data will be taken periodically during normal plant operation to confirm heat transfer capabilities.

9.3.5 SYSTEM RELIABILITY

The CWS is similar to other systems operating in conventional and nuclear power plants. The equipment in this system is designed to applicable codes and standards as listed in Table 9-15. Adequate redundancy, protective devices, and controls are provided to assure reliable and safe operation.

TABLE 9-15

CIRCULATING WATER SYSTEM COMPONENT DESCRIPTION

Traveling Water Screen	
Types	Vertical, single flow through and dual flow through
Quantity	12 (per unit)
Speed (ft/min)	10 (single flow through) 16.5/50 (dual flow through)
Temperature range (F)	0 – 100
Screen Wash Pumps	
Type	Submersible, vertical, centrifugal
Quantity	3 per unit and 2 spares
Capacity each (gpm)	1560
Head (ft)	220
Motor	150 hp, 460 Volt, 3 phase, 60 Hz, 1700 RPM
Codes	National Electrical Manufacturers Association (NEMA), Standards of the Hydraulic Institute, ASME Boiler and Pressure Vessel (B&PV) Codes, Section VIII ANSI B16.5
Trash Trough Wash Pumps	
Type	Submersible, vertical, centrifugal
Quantity	1 per unit
Capacity (gpm)	400
Head (ft)	25
Motor	6.2 hp, 460 Volt, 3 phase, 60 Hz, 1700 RPM
Code	NEMA; Standards of the Hydraulic Institute; ASME B&PV Codes, Section VIII, ANSI B16.5
Circulating Water Pumps	
Type	Vertical, dry pit
Quantity	6 (per unit)
Capacity each (gpm)	200,000
Head (ft)	19.5
Motor	Synchronous 1250 hp, 4160 Volt, 60 Hz, 3 phase, 150 RPM
Codes	NEMA, Standards of the Hydraulic Institute, ASME B&PV Code, Section VIII, Pressure Vessels, ANSI B16.5

9.4 SPENT FUEL POOL COOLING SYSTEM

9.4.1 DESIGN BASIS

The SFPC system is common to both units. The pool contains water with the proper dissolved concentration of boron and has the capacity to store 1830 fuel assemblies.

The SFPC system is designed to remove the maximum decay heat expected from 1613 fuel assemblies, not including a full core off-load. The maximum pool temperature in this case is 120°F. The system is also capable of being used in conjunction with the SDC system to remove the maximum expected decay heat load from 1830 fuel assemblies, including a full core discharge. The maximum SFP temperature in this case is 130°F.

The maximum decay heat load expected from 1613 fuel assemblies, not including a full core off-load, is a function of decay time. For a limiting decay time of 3.5 days, which results in an initial core alteration time of 3.0 days after reactor shutdown, the decay heat load is $22.33x10^6$ Btu/hr. The fuel is assumed to have undergone steady-state burnup at 2738 MWt for an average of 1562.4 days for an 100 assembly batch reload. The total SFP decay heat load as a function of decay time is compared to the heat removal capacity from the two SFP heat exchangers as a function of SRW temperature to show what time after shutdown is acceptable for each SRW temperature condition to maintain the pool at a temperature of 120°F. A maximum SRW temperature of 65°F is required to support a minimum decay time of 3.5 days. In the event that one SFP cooling loop is lost, the remaining loop can remove the heat load while maintaining the pool temperature at 155°F.

The maximum decay heat rate for 1830 fuel assemblies stored in the SFP is a function of decay time. For a limiting decay time of 4.5 days, which results in an initial core alteration time of 3.0 days after reactor shutdown, the decay heat load is $45.96x10^6$ Btu/hr based upon the following hypothetical sequence of events:

1. Eighty-four fuel assemblies are removed from Unit 1 after an average of 1860 days of reactor operation at 2738 MWt, and are replaced with fresh fuel. Unit 1 is then returned to full power.
2. Three-hundred-sixty-five days after the Unit 1 refueling, 84 fuel assemblies are removed from Unit 2 after an average of 1860 days of irradiation and are replaced with fresh fuel. Unit 2 is then returned to full power.
3. Three-hundred-sixty-five days after the Unit 2 refueling, 84 fuel assemblies are removed from Unit 1 after an average of 1860 days of irradiation and are replaced with fresh fuel. Unit 1 is then returned to full power.
4. This refueling cycle continues until the pool contains 1613 fuel assemblies at the end of a Unit 2 refueling. It has been conservatively assumed that the 67 oldest assemblies have been removed from the pool to allow for complete filling of the racks with newer fuel.
5. Unit 1 is then shutdown 60 days after the previous Unit 2 shutdown and the entire core is offloaded after a minimum of 4.5 days of decay. At this point, it is conservatively assumed that the fuel has completed its current cycle, and is therefore at maximum irradiation.

Upon completion of the last operation, the pool will contain 1830 fuel assemblies, with each discharge subjected to different periods of irradiation and decay, in accordance with the table below assuming the minimum decay time of 4.5 days:

	Number of Assemblies	Irradiation Period (Days)	Decay Period (Days)
a.	17	1860	6964.5
b.	84	1860	6599.5
c.	84	1860	6234.5
d.	84	1860	5869.5
e.	84	1860	5504.5
f.	84	1860	5139.5
g.	84	1860	4774.5
h.	84	1860	4409.5
i.	84	1860	4044.5
j.	84	1860	3679.5
k.	84	1860	3314.5
l.	84	1860	2949.5
m.	84	1860	2584.5
n.	84	1860	2219.5
o.	84	1860	1854.5
p.	84	1860	1489.5
q.	84	1860	1124.5
r.	84	1860	759.5
s.	84	1860	394.5
t.	84	1860	64.5
u.	217	1860	4.5

The total SFP decay heat load as a function of decay time is compared to the heat removal capacity from both loops of SFPC as a function of SRW temperature, supplemented with one loop of SDC to show what time after shutdown is acceptable for each SRW temperature condition to maintain the pool at a temperature at 130°F. A maximum SRW temperature of 75°F is required to support a minimum decay time of 4.5 days.

9.4.2 SYSTEM DESCRIPTION

The SFPC System shown in Table 9-16 and Figure 9-7 is a closed-loop system consisting of two half-capacity pumps and two half-capacity heat exchangers in parallel, a bypass filter that removes insoluble particulates, and a bypass demineralizer that removes soluble ions. The SFPC heat exchangers are cooled by service water (SRW).

Skimmers are provided in the SFP to remove accumulated dust from the pool. The clarity and purity of the water in the SFP, refueling pool, and the RWT are further maintained by passing a portion of the flow through the bypass filter and/or demineralizer. The SFP filter and demineralizer removes fission products from the cooling water in the event of a leaking fuel assembly.

Connections are provided for tie-in to the SDC system to provide for additional heat removal in the event that 1830 fuel assemblies are contained in the pool. When the pressure in the SDC system is greater than the design pressure of the SFPC system, the SFPC system is isolated from the SDC system via two manual isolation valves. Although not required by the design code, double valve isolation is provided at this system interface to meet the original FSAR design basis (FCR 90-87).

The entire SFPC system is tornado-protected and is located in a Seismic Category I structure. Borated makeup water comes from the RWT. Non-borated makeup water comes from the demineralized water system.

9.4.3 COMPONENTS

9.4.3.1 Functional Description

A description for the spent fuel pool cooling system is contained in Table 9-16.

9.4.3.2 Codes and Standards

The following codes and standards were used in the design of the SFPC System components:

Pump	Standards of: ASME (III, VIII, IX, PTC8.2), ASTM, NEMA, ANSI
Heat Exchanger	Standards of: Tubular Exchanger Manufacturers Association (TEMA), ASME (III, VIII, IX), ASTM, ANSI
Filter	ASME III C and ASME VIII paragraph UW-2(a)
Ion Exchanger	ASME III C and ASME VIII paragraph UW-2(a)
Valves, Piping, Fittings	ANSI B31.7 Class III

9.4.3.3 Tests and Inspections

Each component is cleaned and inspected before installation and the assembled systems flushed with demineralized water. The flow paths, flow capacity and mechanical operability are tested by operation. The head and capacity of the pumps are also tested.

Instruments are calibrated prior to tests. Alarm functions are checked for operability and limits during preoperational testing. During normal operation, periodic tests will be made to confirm design criteria.

9.4.4 SYSTEM OPERATION AND RELIABILITY

In the normal case (i.e., with no full-core off load), if one SFPC loop is lost, the remaining loop can remove decay heat while maintaining the pool temperature at 155°F. In the case of total loss of SFPC with 1613 fuel assemblies in the pool, it would take more than 8 hours to raise the pool temperature from 155°F to 210°F. The case of total loss of SFP cooling is only discussed to demonstrate the time available to take appropriate action in such an event to preclude boiling, and the resulting loss in pool water level. The design of the SFPC System and pool structural components (e.g., pool liner plate, SFPC piping and pumps) for total loss of cooling is not part of the system's design basis.

The most serious failure to the system is the loss of SFP water. This is avoided by routing all SFP piping connections above the water level and providing them with siphon breakers to prevent gravity drainage.

The SFP is designed to preclude the loss of structural integrity. Section 5.6.1 describes the analysis made to verify that the structural integrity cannot be impaired. Additional design and quality control requirements for the SFP are given in Section 6.3.5.1. However, if a leak from the SFP is postulated, the capabilities for controlling the leak are as follows:

Makeup water can be supplied indefinitely to the SFP at a rate of at least 150 gpm. It can usually be supplied at a greater rate for a period of many days, but this depends upon plant conditions. The makeup water flow path is as follows:

a. Source - Well water

b. Portable Makeup Demineralizers
   - Typical capacity 150 gpm or more

c. Demineralized Water Storage Tank
   - Storage capacity 350,000 gallons

d. Four Reactor Coolant Makeup Pumps (Normally run one per unit)
   - Capacity 165 gpm each, less the amount required for reactor coolant makeup

e. Two RWTs (One per unit)
   - Storage capacity 420,000 gallons
   - Required to have 400,000 gallons during operation
   - During refueling this water has been transferred to the refueling pool where it is also available for pumping if conditions permit

f. Two Spent Fuel Cooling Pumps (One per RWT)
   - Capacity 1390 gpm each

g. Spent Fuel Pool

The two halves of the SFP can be isolated from each other and 830 fuel assemblies, as a minimum, can be stored in the non-leaking half.

The four Emergency Core Cooling System (ECCS) equipment rooms on the lowest level of the Auxiliary Building (Figure 1-5) can be prevented from flooding by shutting their watertight doors. In addition, each ECCS pump room is also drained by an 80 gpm sump pump. The remainder of this level is drained by two sump pumps at a rate of 160 gpm. The sump pumps discharge to the Miscellaneous Waste Processing System (MWPS), which has storage capacity of 8000 gallons and can process 128 gpm.

TABLE 9-16

SPENT FUEL POOL COOLING SYSTEM COMPONENT DESCRIPTION

<u>Pump</u>	
Type	Horizontal, centrifugal with mechanical seals
Number	2
Capacity (each)	1390 gpm
TDH	200 feet
Materials	
Casing	American Society for Testing and Materials (ASTM) A296, Gr CA-15 or ASTM A217, Gr CA-15
Stuffing Box Extension Assy. (Backhead)	ASTM A296, Gr CA-15, ASTM A217, Gr CA-15, ASTM A487 Gr CA-15, or ASTM A487 Gr CA6NM Class A
Motor	100 hp, 460 Volt, 60 Hz, 3 phase, 3550 RPM
<u>Heat Exchanger</u>	
Type	Horizontal counter flow Straight tube rolled and seal welded into tube sheets
Number	2 in parallel
Heat Transfer area (each)	1920 ft^2
Materials	
Shells	C.S. SA-285-C
Tubes	SS-304, SA-213
Tube Sheets	SS-304, SA-240
Shell side relief valve setpoint	150 psig
<u>Fuel Pool Filter</u>	
Type	Cartridge
Number	1
Design/Operating Flow	128/120 gpm
Design Pressure	175 psig
Design Temperature	250°F
Material	ASTM SA240, Type 304
<u>Fuel Pool Demineralizer</u>	
Type	Mixed bed, non-regenerable
Number	1
Design/Operating Flow	128/120 gpm
Design Pressure	200 psig
Design Temperature	250°F
Resin	Mixed (anion, cation)
Materials	ASTM SA240, Type 304

TABLE 9-16

SPENT FUEL POOL COOLING SYSTEM COMPONENT DESCRIPTION

SFP Piping, Fittings, Valves	
Material	Stainless Steel 304
Design Pressure	160 psig
Design Temperature	150°F/155°F[a]
Joints 2-1/2" and Larger	Butt-welded except at flanged equipment
Joints 2" and Smaller	Socket weld except at flanged equipment
Valves 2-1/2" and Larger	Stainless steel, butt weld-ends, 150 psi
Valves 2" and smaller	Stainless steel, socket weld ends, 150 psi
Relief valve setpoint	150 psig (on tube side of spent fuel pool cooling heat exchanger)
Butterflies 3" and larger	Rubber seated carbon steel lug type, 150 psi

[a] Portions of the SFP Cooling System are designed for a maximum postulated temperature of 155°F [Section 9.4.4, Doc. No. 92-769(M601)].

9.5 COOLING WATER SYSTEMS - COMPONENT COOLING, SERVICE WATER, AND SALTWATER

9.5.1 DESIGN BASIS

The CC and SRW systems are designed to remove heat from the plant's various auxiliary systems. The Saltwater System provides the cooling medium for the CC and SRW heat exchangers, and the ECCS pump room air coolers. System components are rated for maximum duty requirements during normal and SDC, and are also capable of providing heat removal during a LOCA. The CC and SRW systems serve as an intermediate barrier between the various auxiliary systems and the saltwater system.

9.5.2 SYSTEM DESCRIPTIONS

9.5.2.1 Component Cooling System

Figures 9-6 (Unit 1) and 9-25 (Unit 2) shows the schematic diagram of the CC. The system for each unit consists of three motor-driven component cooling circulating pumps, two component cooling heat exchangers (Table 9-17), a head tank, associated valves, piping, instrumentation, and controls.

The component cooling heat exchangers are designed for a CC supply temperature of 95°F (a range of 70°F-95°F is acceptable during normal operating conditions), with a saltwater cooling supply temperature of 90°F, at normal operating conditions. Component cooling water may reach as high as 120°F during a LOCA and during plant cooldown and cold shutdowns.

The items cooled by CC include:

a. Letdown heat exchanger
b. Shutdown cooling heat exchangers
c. Miscellaneous waste processing heat exchanger (retired in place)
d. Waste gas compressor aftercoolers and jacket coolers
e. Control element drive mechanism (CEDM) coolers
f. RCP mechanical seals and lube oil coolers
g. LPSI pump seals and coolers
h. HPSI pump seals and coolers
i. Containment penetration cooling
j. Reactor support cooling
k. Steam generator lateral support cooling
l. Coolant waste evaporators (Retired in place)
m. Reactor coolant and miscellaneous waste sampling system
n. Degasifier vacuum pump cooler
o. Post-accident sample system
p. Reactor coolant drain tank heat exchanger

During normal plant operation, one of the pumps and one of the heat exchangers are required for cooling service.

During normal plant cooldowns from 300°F to ≤ 140°F, two CCW pumps and two CCW heat exchangers are required to provide maximum reactor decay heat removal. During post-LOCA long-term core cooling two CCW pumps and two CCW heat exchangers provide the necessary cooling capacity to remove the decay heat from the two shutdown cooling heat exchangers.

The CCW heat exchangers are designed such that, given any single failure, the CC heat exchangers can remove sufficient reactor decay heat to ensure that the containment pressure and temperatures remain within acceptable values during post-LOCA long-term core cooling. Because the two CCW system trains are cross- connected, there are certain failures scenarios where CCW flow may be directed through a CCW heat exchanger which is not removing heat (e.g., the CCW heat exchanger has lost saltwater cooling flow). In these cases CCW system heat removal performance is enhanced by isolating CCW flow to the non-heat removing CCW heat exchanger and directing all CCW flow through the in-service CCW heat exchanger. Depending on the failure, it may be required to isolate the non-functioning CCW heat exchangers to ensure that the post-LOCA containment pressures and temperatures remain within acceptable values.

The CC pump motors are supplied from two separate 480 Volt engineered safety feature (ESF) busses, with the third motor having two breakers, one from each bus. If a loss of offsite power occurs, the pumps can be supplied by the Emergency Diesel Generators (EDGs). During normal shutdown cooling, two pumps are running with the third pump on standby. Low discharge header pressure is annunciated in the Control Room where the operator can start the third pump.

A head tank allows for expansion of the system water and provides sufficient net positive suction head (NPSH) for the component cooling circulating pumps. Makeup can be added to the system to maintain head tank level. The source of makeup water is the plant demineralized water system. Additional makeup capacity may be provided from the condensate system.

A chemical additive tank connected to the system permits maintenance of the proper corrosion inhibitor concentration in the CC.

The operation of each system is controlled and monitored in the Control Room with the following instrumentation:

a. Temperature indicators and high temperature alarms from the component cooling heat exchangers and RCP CC outlets;
b. Temperature indicators on the shutdown cooling heat exchangers;
c. Pressure indicators and low pressure alarms for each discharge header;
d. Level indicators and high-low level alarms for the head tank;
e. Handswitches and indicating lights for the pumps and remotely-operated control valves;
f. Radiation indicators and high radiation level alarm from the discharge side to the suction side of the CC pumps; and,
g. Low component cooling flow alarm to RCPs.

9.5.2.2 Service Water System

The SRW System as shown in Figures 9-9 (Unit 1) and 9-27 (Unit 2) is a closed system and uses plant demineralized water with a corrosion inhibitor added. The system removes heat from turbine plant components, blowdown recovery heat exchangers, containment cooling units, SFPC heat exchangers, AFW Pump Room Emergency Cooling Fan Coil Units, and Fairbanks Morse Emergency Diesel Generator heat exchangers.

The system has been divided into two subsystems in the Auxiliary Building to meet single failure criteria. Each subsystem has a head tank to maintain the subsystem's pressure and to allow for thermal expansion. Demineralized water makeup to the head tank is automatically controlled by level controllers. Additional makeup capacity may be provided from the condensate system.

Operating instructions provide the operators with procedures for aligning alternate sources of SRW make-up during accident or abnormal operating conditions. A cross-connection (via temporary hose) between the Saltwater and SRW Systems could be established if all non-seismic make-up water sources (demineralized water, condensate, or fire system) are unavailable.

The SRW additive tank is connected to both subsystems to allow chemical addition and control to prevent corrosion.

During normal operation, both subsystems are required and are independent to the degree necessary to assure the safe operation and shutdown of the plant-assuming a single failure. During the shutdown, operation of the SRW system is the same as normal operation, except that the heat loads are reduced.

During LOCA operation, each of the two subsystems for the two nuclear units will cool a maximum of two containment air coolers and one diesel generator. Although Unit 2 has identical heat loads and flow requirements for LOCA operations, Unit 1 subsystems do not have identical heat loads as Unit 1 has only one service water-cooled diesel generator. Service Water Subsystem 12 cools Diesel Generator 1B, and 1A is cooled from an independent cooling source located in the safety-related Diesel Generator Building. The original design heat removal capability of three of the four containment cooling units was to provide the same heat removal capability as the containment spray system. The analysis of these systems operating together post-LOCA in accordance with the Technical Specification requirements is presented in Section 14.20.

There are three SRW pumps in all. Each of two pumps is powered from a different ESFs 4 kV bus. The third pump is capable of being powered from either ESFs 4 kV bus. In the event that one bus is unavailable, a manual transfer capability to the operating bus is provided for this pump.

A low discharge header pressure will annunciate in the Control Room and the operator can then manually activate the standby pump.

The turbine plant components cooled by SRW include:

a. Generator isolated 3 phase bus duct coolers
b. Exciter air coolers
c. Generator hydrogen coolers
d. Stator liquid coolers (Unit 1 only)
e. Circ. Water System Priming Pump seal water coolers
f. Condenser vacuum pump seal water coolers
g. Feed pump turbine lube oil coolers
h. Condensate booster pump lube oil and seal water coolers
i. Instrument and plant air compressors and aftercoolers
j. Turbine lube oil cooler
k. Electro-hydraulic oil coolers

l. Turbine Building sample cooling system
m. Seal oil system coolers (Unit 2 only)
n. Auxiliary Feed Pump Room Air Cooler

Service water to the Turbine Building is not automatically isolated upon a seismic event. However, to ensure that this portion of the system will perform its pressure boundary function and provide continued cooling to the diesel generators during and after a seismic event, the entire Turbine Building SRW piping was walked down and evaluated for seismic adequacy. Consequently, a few small bore pipes and associated supports were modified to protect the non-safety-related portion of the system from potential seismically-induced spatial interaction with adjacent stationary structures and/or pipes and components.

Supply and return line redundancy is provided for containment cooling units, and diesel generators. Redundancy for SFPC is provided by cooling one SFP cooler from each unit.

Radiation monitors are installed in the SRW return header from the SFP coolers to detect possible in-leakage of radioactive liquids through the heat exchangers (Section 11.2.3.1).

9.5.2.3 Saltwater System

The Saltwater System has three pumps for each unit æ Nos. 11, 12, 13 in Unit 1, and Nos. 21, 22, 23 in Unit 2. The pumps provide the driving head to move saltwater from the intake structure, through the system and back to the circulating water discharge conduits (Figures 9-8 and 9-26). The system is designed such that each pump has sufficient head and capacity to provide cooling water for the SRW and CC Systems, as required by 10 CFR Part 50, Appendix A. The system also cools the ECCS pump room air coolers. The maximum recommended pump flow for each pump is 25,000 gpm. Although under most conditions this is not a limiting feature, when storm conditions consisting of the lowest expected tide of 4’0” below mean sea level and the lowest expected barometric pressure of 26.9” of mercury are considered, sufficient net positive suction head may not be available at flows above 25,000 gpm.

Power is supplied to Pumps No. 11 and 21 by 4 kV Busses No. 11 and 21, respectively, and Pumps No. 12 and 22 from 4kV Busses No. 14 and 24, respectively. Pumps No. 13 and 23 can receive power from either of the 4 kV busses in their respective units. (Figure 8-1) Pumps No. 11, 12, 21 and 22 start automatically on a SIAS or Shutdown Sequencer Signal. Pump No. 13(23) is aligned to back up Pump Nos. 11 or 12 (21 or 22). Pump No. 13(23) starts on a SIAS on shutdown sequencer signal when the backed up pump [Nos. 11 or 12 (21 or 22)] fails to start. A low discharge header pressure alarm will annunciate in the Control Room where the operator can manually activate the standby pump. The motors and controls for the saltwater pumps are located at or above Elevation +17’00” to protect them against flooding. The peak hypothetical tide and storm surge is 16.2’00” above mean low water. (Sections 7.3.2.2 and 7.3.2.3)

The Saltwater System consists of two subsystems in each unit. Each subsystem provides saltwater to two SRW heat exchanger, a CC heat exchanger and the ECCS pump room air cooler in order to transfer heat from those systems to the Chesapeake Bay. Seal water for the circulating water pumps is supplied by both subsystems. A self-cleaning strainer is installed upstream of each SRW heat exchanger.

During normal operation, both subsystems in each unit are in operation with one pump running on each header and a third pump in standby. If needed, the standby pumps can be lined-up to either supply header. Normally, the saltwater flow through the CC heat exchangers is throttled and the SRW heat exchanger saltwater valves are full open to provide sufficient cooling to the heat exchangers, while maintaining total subsystem flow below the maximum recommended value to prevent pump runout.

The operator has the option to reduce saltwater flow to the SRW heat exchangers by placing the SRW heat exchanger saltwater outlet valve flow controllers in automatic to maintain saltwater flow to each plate heat exchanger at a nominal value of 4550 gpm. At the design saltwater flow rates, the SRW heat exchangers can remove the accident heat load at saltwater inlet temperatures up to 90°F.

The saltwater pumps were originally designed for a nominal flow of 20,000 gpm with a minimum flow requirement of 10,000 gpm. To allow system operation in lower flow configurations, a saltwater bypass line exists around the SRW heat exchangers. The saltwater bypass valves are normally shut; however, they may be automatically throttled by a pressure controller to maintain saltwater header pressure within selected limits.

Operation following a LOCA has two phases æ before the RAS and after the RAS. One subsystem can satisfy the cooling requirements of both phases.

After a LOCA, but before an RAS, each subsystem will cool two SRW heat exchangers and an ECCS pump room air cooler. Any flow established to the CC heat exchanger prior to the accident will continue during this phase. The minimum required saltwater flow is 4,000 gpm to each SRW heat exchanger, and 400 gpm to each ECCS pump room air cooler at 90°F. There is no required flow to the CC heat exchangers. The SRW heat exchanger saltwater outlet valves will remain full open or, if the outlet valves are in automatic, the saltwater flow controllers will continue to maintain flow at the same setpoint used during normal operation.

When an RAS occurs, the minimum required flow to each SRW heat exchanger remains at 4,000 gpm, and each ECCS pump room air cooler remains at 400 gpm with saltwater temperatures at 90°F. Flow is initiated or increased to the CC heat exchangers at a minimum required flow of 5,500 gpm each. The operator will throttle saltwater flow through the CC heat exchangers to maintain CC temperature. If in use to meet saltwater pump minimum flow requirements, the SRW heat exchanger bypass control valve is automatically throttled by the pressure controller to maintain the saltwater header pressure within the selected limits.

Should a piping rupture or blockage occur downstream of the heat exchangers and air coolers, an alternate flow path may be employed so the function of the components will not be impaired.

In an accident situation, control air for the throttling valves is supplied by two 64 scfm (Unit 1)/64 scfm (Unit 2), Seismic Category I air compressors. These designated air compressors are used because the Instrument Air System compressors which normally supply control air to the valves are not designated safety-related, and are not required to be operational after an accident. The compressors are normally not running, but will start automatically on receipt of a SIAS, or may be manually started from the Control Room. Upon evacuation of the

Control Room, remote manual control may be shifted to local manual control, and SIAS input to the compressors is overridden.

The throttling system meets all applicable requirements of IEEE 279.

The Saltwater Chemical Addition System serves both the Unit 1 and Unit 2 Saltwater Systems to minimize the marine fouling of piping and heat exchanger surfaces. This system has the ability to inject approved chemicals into each saltwater header, as necessary.

9.5.3 TESTING AND INSPECTION

Each component was inspected and cleaned prior to installation into the system.

Instruments were calibrated during testing. Automatic controls were tested for actuation at the proper setpoints. Alarm functions and limits were checked for operability during preoperational testing. The safety valves were set and checked.

Figures 9-6, 9-8, 9-9, 9-25, 9-26, and 9-27 show the CC, saltwater cooling, and SRW systems.

The pre-operational testing verified the following:

a. Pumps produce proper capacity and discharge head with one, two or more pumps.
b. System components receive proper flow for all modes of operation (i.e., normal, shutdown and LOCA).
c. Instrumentation and controls are functioning or responding properly.
d. Motor-operated (MOV) and control valves function.

Data is taken periodically during normal plant operation to confirm heat transfer capabilities.

9.5.4 RATINGS AND CONSTRUCTION OF COMPONENTS

Components of the cooling water system are described in Table 9-17.

9.5.5 SINGLE FAILURE ANALYSIS

The results of a single failure analysis (Table 9-17A) show that no single active failure at any time nor any single passive failure after recirculation from the containment sump will prevent the safety feature systems from fulfilling their design function.

The Nuclear Regulatory Commission (NRC) approved the application of a revised methodology for the evaluation of passive failures in moderate energy systems (Reference 1). The revised methodology assumes the passive failure to be a through-wall leakage crack of dimensions equal to one-half the pipe diameter in length, and one-half the wall thickness in width. The passive failure is postulated to occur in the largest pipe in the area to be evaluated, at least 24 hours after the initiating event. This methodology was specifically evaluated for a passive failure of the Saltwater System piping in the SRW Pump Room, but may be adopted for other moderate energy systems if supported by a similar analysis to that performed on the Saltwater System to ensure the validity of the revised methodology for those systems/subsystems. Systems evaluated by the revised methodology are annotated as such.

9.5.6 REFERENCES

1. Letter from D. G. McDonald, Jr. (NRC) to R. E. Denton (BGE), dated February 24, 1995, Methodology for Postulating Passive Failure Pipe Breaks

TABLE 9-16A

HEAT EXCHANGER CONTROL VALVE POSITION

		NORMAL	LOCA BEFORE RECIRCULATION	LOCA DURING RECIRCULATION	ALTERNATE MODE
Heat Exchanger Discharge Control Valve					
Component Cooling	CV-5206	Throttle[a]	Throttle	Throttle	Closed
	CV-5208	Throttle[a]	Throttle	Throttle	Closed
	CV-5163	Open	Open	Open	Closed
	CV-5165	Closed	Closed	Closed	Open
	CV-5166	Closed	Closed	Closed	Open
Service Water	CV-5209	Open[f]	Open[f]	Open[f]	Open[f]
	CV-5210	Open[f]	Open[f]	Open[f]	Open[f]
	CV-5211	Open[f]	Open[f]	Open[f]	Open[f]
	CV-5212	Open[f]	Open[f]	Open[f]	Open[f]
	CV-5153	Open	Open	Open	Closed
	CV-5155	Closed	Closed	Closed	Open
	CV-5156	Closed	Closed	Closed	Open
Emergency Core Cooling	CV-5171	Closed[b]	Closed[b]	Closed[b]	Closed
	CV-5174	Open	Open	Open	Closed
	CV-5175	Open	Open	Open	Closed
	CV-5177	Closed	Closed	Closed	Open
	CV-5178	Closed	Closed	Closed	Open
Heat Exchanger Inlet Control Valve					
Component Cooling	CV-5160	Open[a]	Open	Open	Closed
	CV-5162	Open[a]	Open	Open	Open
Emergency Core Cooling	CV-5170	Closed[b][c]	Closed[b][c]	Closed[b][c]	Closed[c]
	CV-5173	Closed[b]	Closed[b]	Closed[b]	Closed[b]
Service Water	CV-5150	Open	Open	Open	Closed
	CV-5152	Open	Open	Open	Open
Heat Exchanger Bypass Control Valve					
Service Water	CV-5154	Closed[g]	Closed[g]	Closed[g]	Closed
	CV-5157	Closed[g]	Closed[g]	Closed[g]	Closed[g]
Saltwater Strainer Control Valve					
Diverter Valve	CV-5148	Open[d]	Open[d]	Open[d]	Open[d]
	CV-5151	Open[d]	Open[d]	Open[d]	Open[d]
	CV-5158	Open[d]	Open[d]	Open[d]	Open[d]
	CV-5159	Open[d]	Open[d]	Open[d]	Open[d]
Flushing Valve	CV-5148A	Closed[e]	Closed[e]	Closed[e]	Closed[e]
	CV-5151A	Closed[e]	Closed[e]	Closed[e]	Closed[e]
	CV-5158A	Closed[e]	Closed[e]	Closed[e]	Closed[e]
	CV-5159A	Closed[e]	Closed[e]	Closed[e]	Closed[e]

TABLE 9-16A

HEAT EXCHANGER CONTROL VALVE POSITION

(a) Routinely only one CC heat exchanger is required for heat removal during normal operations. Saltwater flow to the other CC heat exchanger may be secured.

(b) ECCS pump room air cooler saltwater valves are automatically opened in order to regulate the ECCS pump room ambient temperature.

(c) A bypass line has been installed around this valve to allow for fluid expansion back into the saltwater header.

(d) Diverter valve is normally open; closes during strainer flush.

(e) Flushing valve is normally closed; opens during strainer flush.

(f) The SRW plate heat exchanger outlet valves are normally full open. They may be throttled and controlled by an FIC if the operator needs to reduce saltwater flow.

(g) Bypass valve is normally shut. It may be placed in automatic to assist in satisfying pump minimum flow requirements.

TABLE 9-16B

SALTWATER SYSTEM AIR COMPRESSORS

Type	Oil-less, Reciprocating Duplex (each SWAC has two compressor units mounted on a common air receiver tank)
No. of Stages	One
Quantity	Two
Design Capacity (scfm)	64
Design Pressure (psig)	100
Motor	Electric Motors, 10 hp each, 460 Volt, 3 phase, 60 Hz (two motors per SWAC)
Accessories	Air Receiver, Air-cooled Aftercooler, Automatic Condensate Trap
Seismic Requirements	Category I
Codes	Receiver - ASME Section VIII, Motor - NEMA

TABLE 9-17

COOLING SYSTEM COMPONENT DESCRIPTION

Component Cooling Pumps

Type	Centrifugal, horizontal, double volute, with mechanical seal
Quantity	3
Capacity each (gpm)[c]	5000
Head (feet)[c]	100
Material	
Case	ASTM A216-59T-WCB
Impeller	ASTM B145, Gr 4A ASTM B584 C83600, C87500, or C87600
Shaft	ASTM A276, Type 410 ASTM A276, Type 316 (ALT)
Motor	150 hp, 480 Volt, 60 Hz, 3 phase, 1750 RPM
Codes	Motor: NEMA Pump: Standards of the Hydraulic Institute, ASME VIII and IX

Component Cooling Heat Exchangers

Type	Horizontal, counterflow, straight tubes rolled into tubesheets
Quantity	2
Design duty each (Btu/hr)	10.4×10^6 (Normal)[a] 122×10^6 (3.5 hrs after shutdown)[a] 31.2×10^6 (27.5 hrs after shutdown) [a] 43.5×10^6 (long term cooling following a LOCA) [a]
Heat transfer area, each (ft^2)	5860
Design pressure (psig)	Shell side: 150 Tube side: 50
Design temperature (°F)	Shell side: 200 Tube side: 200
Material	
Shell	Carbon steel ASTM A285, Gr C
Tubes	90-10 Cu-Ni ASTM B111
Tube Sheets	Aluminum bronze ASTM B171-67
Codes	ASME Section VIII, TEMA Class R

Head Tank

Type	Horizontal
Quantity	1
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Volume (gallons)	2550
Material	
Shell	ASTM A455A
Dished head	ASTM A455B
Code	ASME Section VIII

Additive Tank

Type	Vertical
Quantity	1
Design pressure (psig)	150
Design temperature (°F)	200
Volume (gallons)	75
Material	Carbon steel
Code	ASME Section VIII

TABLE 9-17

COOLING SYSTEM COMPONENT DESCRIPTION

Component Cooling Piping, Fittings, and Valves

Piping material	Carbon steel, seamless
Design pressure (psig)	150
Design temperature (°F)	180
Construction:	
2-1/2" and larger	
a. Gate and globe	Carbon steel, butt weld ends, ANSI 150 psi
b. Check and butterfly	Carbon steel, wafer type, ANSI 150 psi
2" and smaller	Carbon steel, socket weld ends, ANSI 600 psi
Codes	ANSI B31.1 except penetration piping. Penetration piping is designed and fabricated to ANSI B31.7, Class II

Service Water Heat Exchanger

Type	One pass plate and frame
Quantity	4
Capacity each (Btu/hr)	18×10^6 (normal operation)[a] 137×10^6 (LOCA operation)[a,b]
Heat Transfer Area each (ft^2)	7704.4
Design Pressure (psig)	150
Design Temperature (°F)	300
Material	
Pressure Plates	Steel - SA516-70
Plates	Titanium - SB265 GR 1
Port Liners	Titanium - SB337 GR 2 (Saltwater) Stainless Steel - SA312-316 (SRW)
Gaskets	EPDM
Codes	ASME B&PV Code, Section VIII - Pressure Vessels, Section IX - Welding Qualifications, ASTM

Service Water Pumps

Type	Centrifugal, horizontal, double volute, with packed seal
Quantity	3
Capacity each (gpm)[c]	7,050
Head (ft)[c]	180
Material	
Case	ASTM A216, WCA
Impeller	ASTM B145-52, Gr-4A or B584 UNS # C83600 or B584 UNS # C87600
Shaft	ASTM A276, Type 410 or ASTM A276, Type 316
Motor	450 hp, 4000 Volt, 3 phase, 60 Hz 585 RPM
Codes	NEMA, Standards of the Hydraulic Institute, ASTM, ANSI B16.5

TABLE 9-17

COOLING SYSTEM COMPONENT DESCRIPTION

<u>Service Water Head Tank</u>

Type	Vertical
Quantity	2
Design Pressure (psig)	15
Design Temperature (°F)	150
Volume (gallons)	2,350
Material	ASTM A455, Gr A
Code	ASME Section VIII

<u>Service Water Additive Tank</u>

Type	Vertical
Quantity	1
Design Pressure (psig)	175
Design Temperature (°F)	200
Volume (gallons)	75
Material	ASTM A283, Gr C
Code	ASME Section VIII

<u>Saltwater Pumps</u>

Type	Vertical, dry pit
Quantity	3
Design Capacity each (gpm)[c]	15,500
Design Head (ft)[c]	82
Material	
Volute	2% Ni ASTM A48, C1.35 or A439 Type D3
Impeller	ASTM B148, Gr 9D or UNS # C95500
Shaft	AISI-C1141 or ASTM A322, Gr 4140
Motor	450 hp, 4000 Volt, 3 phase, 60 Hz, 600 RPM (nominal)
Codes	Motor: NEMA Pump: Standards of Hydraulic Institute, ASME B&PV Code, Section VIII, Pressure Vessels and IX, Welding

<u>Saltwater Strainers</u>

Type	Self-cleaning basket
Quantity	4
Design pressure (psig)	50
Design Temperature (°F)	100
Material	
Body	A416-60
Basket	A240 GR TP316
Code	ANSI B31.1

[a] Per vendor rating sheet; actual heat duty will vary with flow and temperature.

[b] Per accident analysis, no rating sheet available for LOCA; actual heat duty will vary with flow and temperature.

[c] These numbers, together, represent a single point on the pumps' performance curve.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

ITEM	COMPONENT	NO. INSTALLED PER UNIT	NO. NEEDED FOR NORMAL OPERATION	MINIMUM NO. NEEDED FOR OPERATION FOLLOWING LOCA	DESIGN FUNCTION OF COMPONENT
Saltwater	Saltwater Pumps	3	2	1	Provide cooling water for SRW and component cooling heat exchangers and the ECCS pump room coolers.
Saltwater	Service Water Heat Exchangers	4	(a)	(b)	Provide cooling for turbine auxiliaries, SFP coolers, blowdown recovery system, containment coolers, and diesel generators.
Saltwater	Component Cooling Heat Exchangers	2	1	(b)	Provide cooling for reactor auxiliaries, HPSI pumps, LPSI pumps, and SDC heat exchangers.
Saltwater	ECCS Room Coolers	2	-	1	Maintain design temperature in ECCS rooms for long-term operation of the safety feature pumps.
Service Water	Service Water Pumps	3	2	1	Provides driving force for SRW system.
Service Water	Containment Coolers	4	(f)	(b)	Cools the containment.
Service Water	Diesel Generators	2(c)	-	1	Provides source of emergency on-site power.
Component Cooling	Component Cooling Water Pumps	3	(d)	1	Provides driving force for CC.
Component Cooling	Low Pressure Safety Injection Pumps	2	(d)	(e)	Provides safety injection water and SDC.
Component Cooling	High Pressure Safety Injection Pumps	3	-	1	Provides safety injection water.
Component Cooling	Shutdown Cooling Heat Exchanger	2	(d)	(b)	Provides cooling medium for spray water to remove heat from the containment following recirculation and SDC.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

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[a] Four SRW heat exchangers are needed.

If one saltwater subsystem is out for maintenance, the two subsystems of the SRW system may be cross-connected and the two remaining heat exchangers utilized to remove the heat load during normal operations. The two subsystems are physically separated during accident conditions; only the diesel generator connected to the operable SRW heat exchanger will be considered operable. Refer to Note (c).

[b] Each containment cooler was originally designed to remove 1/3 of the containment design heat load and each containment spray pump-shutdown heat exchanger was originally designed to remove 1/2 of the containment design heat load. The original design heat removal capability of three of the four cooling units was to provide the same heat removal capability as the containment spray system. The analysis of these systems operating together post-LOCA in accordance with the Technical Specification requirements is presented in Section 14.20. There are several combinations of equipment that could be utilized to remove heat from the containment. Each SIAS channel would actuate two containment coolers and one spray pump. For each shutdown cooling heat exchanger, one component cooling heat exchanger would be placed in service. If three containment coolers are utilized, at least three SRW heat exchangers need to be placed in service.

[c] Two diesel generators are installed per unit; however, only one diesel generator for Unit 1 and both diesel generators for Unit 2 are served by the SRW System.

[d] The LPSI pumps, the SDC heat exchangers and the CC System (i.e., CC heat exchangers and CC pumps) provide heat removal for a normal plant cooldown. One LPSI pump, one SDC heat exchanger, one CC pump, and one CC heat exchanger would provide cooldown at a slower rate. However, even at this slower rate, the plant is maintained in a safe condition (Sections 6.3.2.5, 9.5.2.1).

[e] One LPSI pump is required when suction is from the RWT and none is required when suction is switched to the containment sump during the recirculation mode of cooling.

[f] Three containment air coolers are normally in operation. Occasionally, during extended periods of high outside temperatures, all four coolers are used to limit average containment temperature to 120°F.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

ACTIVE FAILURES

SYSTEM	FIGURE	COMPONENT	TYPE OF FAILURE	CONSEQUENCES
Saltwater	9-8	Remotely actuated valves (CV-5150, 5152, 5153, 5155, 5156, 5160, 5162, 5163, 5165, 5166)	Fails to open or close, as applicable	These values are only operated to align the redundant discharge header. A failure of any valve would not impair the integrity of the system or prevent it from functioning.
Saltwater	9-8	Remotely actuated valves (CV-5209, 5210, 5211, 5212)	Fails open (fails to throttle)	Normally, these valves are full open. Should any one of these valves fail open while in automatic, saltwater flow to the associated PHE will be increased, improving the component's heat removal capability. The other PHE on the subsystem would continue to operate. Total system flow will increase or, if the saltwater bypass valve is in automatic, will be automatically adjusted by the bypass line CVs. The other saltwater subsystem would be unaffected by this failure and remain capable of removing the full design accident heat load.
Saltwater	9-8	Remotely actuated valves (CV-5154, 5157)	Fails closed (fails to throttle)	Normally, the saltwater bypass valves will be shut. However, should a bypass valve fail closed while in automatic, the PHE saltwater flow will increase or, if the FIC is in automatic, the outlet throttle CVs will maintain the flow through the PHEs at the setpoint. However, if only the SRW PHEs are in service, and the PHE saltwater outlet valves are in automatic, saltwater flow may drop below the minimum required flow for pump operation. The safety-related functions of the saltwater and SRW systems would not be immediately impacted. The operator can raise flow, if desired, by manually raising the FIC setpoint or remotely opening the PHE outlet valves, disabling the FIC. The other saltwater subsystem would be unaffected and capable of removing the full design accident heat load.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

ACTIVE FAILURES

SYSTEM	FIGURE	COMPONENT	TYPE OF FAILURE	CONSEQUENCES
Saltwater	9-8	Strainer	Basket clogs	The strainer is designed to flush automatically and on manual initiation. Should clogging occur, the affected strainer will eventually reach its dP limit and alarm setpoint. Heat exchanger saltwater flow will be maintained by the FIC, if in automatic, or will start to gradually decrease. Eventually the saltwater low flow alarm setpoint would be reached. A handhole allows quick inspection and manual cleaning. The affected strainer can be deenergized, allowing the unaffected strainer to resume automatic flushing. The other saltwater and SRW subsystems would be unaffected and capable of removing the full design accident heat load.
Saltwater	9-8	Strainer flushing valves (CV-5148A, 5151A, 5158A, 5159A)	Fails to cycle properly	If the valves fail to shut, the affected strainer would continue to flush and remain relatively clean. Condition would initiate system trouble alarm to alert operator. Without operator action, the interlock between the two subsystem strainers will prevent flushing of the unaffected strainer. As the strainer clogs, PHE saltwater flow will gradually decrease or, if in automatic, the FIC will compensate to maintain minimum flow to the heat exchanger. The operator can deenergize the failed strainer to allow the unaffected strainer to resume its automatic flushing sequence. Both PHEs will continue to remove their design basis heat load until the heat exchanger low flow setpoint is reached. If the valves fail to open, the operator will be alerted by the system trouble alarm. The affected strainer will gradually clog. Saltwater flow to the associated PHE will start to decrease. (Initially, the associated heat exchanger FIC will compensate, if in automatic.) The flushing circuit on the unaffected strainer would continue to function. Both PHEs will continue to remove their design basis heat load until the heat exchanger low flow setpoint is reached on the affected side.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

ACTIVE FAILURES

SYSTEM	FIGURE	COMPONENT	TYPE OF FAILURE	CONSEQUENCES
				The other saltwater and SRW subsystems would be unaffected and capable of removing the full design accident heat load.
Saltwater	9-8	Strainer diverter valves (CV-5148, 5151, 5158, 5159)	Fails to cycle properly	If the valve fails to shut during regeneration, the saltwater trouble alarm will be activated. This failure would lead to less effective flushes, probably resulting in an increased number of automatically-initiated flushes. This would eventually have the same effect as a flushing valve failing closed. If the valve fails to open during the flush cycle, the saltwater trouble alarm will be activated. The number of automatic strainer flushes would increase. Eventually this would have the same affect as a flushing valve failing closed. The other saltwater and SRW subsystems would be unaffected and capable of removing the full design accident heat load.
Service Water	9-9B	Valves No. 1, 3, 9, or 11	Fails to close on SIAS	Valves No. 2, 4, 10, and 12 are actuated by a redundant channel and would shut, isolating SRW as required.
Service Water	9-9B	Valves No. 5, 7	Fails to close on CSAS	Valves No. 6 and 8 are actuated by a redundant channel and would shut, isolating SRW as required.
Service Water	9-9B	Valve 27(28)	Fails to close on CSAS	Failure of valve 27(28) could render subsystem 11(21) inoperable. However, subsystem 12(22) would continue to provide the necessary cooling for Unit 1 (Unit 2).
Service Water	9-9B	Check Valves No. 17, 18, 19, 20, 21 or 22	Fails to close under reverse flow	Since in all cases two check valves are provided in series, the second valve would close providing isolation.

NOTE: As shown above, sufficient numbers of all other active components are supplied to provide sufficient redundancy for all modes of operation.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

PASSIVE FAILURE DURING CONTAINMENT SUMP RECIRCULATION

SYSTEM	FIGURE	LOCATION OF RUPTURE	CONSEQUENCES
Saltwater	9-8	Anywhere	Water is lost from one of the two subsystems. Either subsystem can provide all necessary cooling water. Double valves are provided whenever subsystems are tied together. Both of these valves are normally closed. Hence, a rupture of any one valve will not cause failure of both subsystems.
Service Water	9-9B	Valves No. 23, 24, 25, or 26	One subsystem from each unit would be drained and rendered inoperable. However, one subsystem in each unit would continue to operate. This is adequate to provide the necessary cooling for each unit. No single rupture in any location could cause the loss of both subsystems of a unit as two normally closed valves are provided where two subsystems are tied together.
Component Cooling	9-6	Anywhere	The entire system would be lost. The unit can still be maintained in a safe condition since the containment coolers would be utilized in lieu of the spray pumps/shutdown heat exchangers to cool the containment and one of the air cooled spray pumps would be manually aligned from outside the ECCS rooms for safety injection. The HPSI pumps can operate for a minimum of two hrs without cooling water and this is considered sufficient to realign the valves. Flow of one spray pump is sufficient to keep the core covered during the recirculation of the containment sump.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

FLOODING DUE TO A PASSIVE FAILURE

STRUCTURE FLOODED	INDICATION IN CONTROL ROOM	SYSTEM RUPTURED	CONSEQUENCES
Intake Structure	High level alarm/Circulating Water Pumps Trip	Saltwater	The bottom of the Intake Structure is at Elevation 3'. The operator enters the Intake Structure from the Turbine Building at Elevation 12' and the saltwater pump motors are at Elevation 17'. It would take approximately 82 minutes for the water level to reach the motors and approximately 53 minutes to reach the entrance from the Turbine Building. This is sufficient time to allow shutting down one saltwater pump at a time until the leakage stops as visually determined by an operator in the Intake Structure.
Intake Structure	High level alarm/Circulating Pump Trip	Circulating Water	Before the saltwater pump motors would be flooded, the circulating water pump motors would flood and trip, eliminating the source of flooding. In addition, high level switches would trip the circulating water pump motors, eliminating the source of flooding.
Service Water Room	High level alarm in the room with normal service water head tank level	Saltwater[a]	Operators would have sufficient time to identify and isolate the break in the Saltwater System before safety-related equipment required to function would be affected by the break. For a saltwater line break that is limited to a single train, approximately 30 minutes is available to identify the affected train and isolate it. For a saltwater line break in the common portion of the SRW heat exchanger discharge piping, approximately 80 minutes is available to shift to overboard discharge after isolating the break.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

FLOODING DUE TO A PASSIVE FAILURE

STRUCTURE FLOODED	INDICATION IN CONTROL ROOM	SYSTEM RUPTURED	CONSEQUENCES
Service Water Room	High level alarm in room and low level from either SRW level tank	Service Water	One subsystem would be drained. However, the other subsystem would continue to operate and is sufficient to provide all necessary SRW. The entire contents of one SRW subsystem would not flood out the SRW pumps and motors.
Component Cooling Room	High level alarm in room with normal head tank level	Saltwater	Since this room is open to the entire Elevation 5', flooding is not considered credible. A 6" curb is provided at the doorway to provide a room level indication. Flooding would be terminated by closing the remote manual valves.
Containment	Low level alarm from either SRW head tank	Service Water	In the event of a line break associated with any one containment air cooler after the LOCA, it is assumed, as an upper limit, that one subsystem of SRW leaks into containment. The leak volume from one subsystem is approximately 16,000 gallons. Boron dilution, therefore, would be negligible, because the total volume of borated water in the containment structure is in excess of 400,000 gallons.
Component Cooling Room	High level alarm in room with low head tank level	Component Cooling	Since this room is open to the entire Elevation 5', flooding is not considered credible. A 6" curb is provided at the doorway to provide a room level indication.
ECCS Room	High level alarm in room	Safety injection containment spray, containment cooling, or salt water	ECCS room is isolated. Each room is watertight and fully redundant to the other. Remote manual valves would be closed to prevent further flooding.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

FLOODING DUE TO A PASSIVE FAILURE

STRUCTURE FLOODED	INDICATION IN CONTROL ROOM	SYSTEM RUPTURED	CONSEQUENCES
Condenser Pit	High level alarm in Condenser Pit	Circulating Water Expansion Joint	The maximum flood height from an expansion joint rupture in the Turbine Building is the 15'-8" elevation, if the circulating water flow path was not stopped by operator action. The condenser pit would flood and overflow into the Turbine Building. The AFW pumps with local control, SRW pumps and intake structure are protected by watertight doors. The turbine-driven AFW pumps are also protected by drain isolation valves. It would take approximately 45 minutes to reach the watertight doors at elevation 12'-6" and greater than 60 minutes to impact the turbine-driven AFW pumps at an elevation of 13'-3". A Turbine Building flood event requires 37 minutes to reach a height of less than 12'-3". This allows sufficient time for operator action to stop the event.

[a] The passive failure is evaluated using the methodology approved by the NRC in the Safety Evaluation Report dated February 24, 1995. The passive failure is assumed to be a through wall leakage crack of dimensions equal to one-half the pipe diameter in length, and one-half the pipe wall thickness in width. The passive failure is assumed to occur in the largest pipe in the area to be evaluated, at least 24 hours after the initiating event.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

FLOODING DUE TO A PASSIVE FAILURE

NOTE: Power cables to the saltwater pump motors could be submerged by the flooding under certain conditions. The following precautions have been taken to prevent this flooding from causing a failure of the saltwater pump motor cables:

1. These 5 kV Kerite HT and HV insulation type cables are suitable for submerged operation.
   a. The Kerite Company states that saltwater in contact with their 5 kV NS jacketed cables would cause no deleterious effects whatsoever.
   b. The Kerite Company has made numerous documented tests to prove the reliability of their Kerite HT and HV insulation cables in submerged applications. The reliability of this type of cable has also been proven through experience. The Kerite Company has been making cables with Kerite type insulation for over 100 years and has supplied many miles of this type cable for continuously submerged use.
   c. Baltimore Gas and Electric Company has used more than 100,000' of three-phase Kerite medium voltage (2.4 kV, 4 kV and 13 kV) cable for hundreds of circuits at 16 generating units. These cables have been installed for up to 31 years and over half of them are intermittently flooded by fresh or brackish water. This experience totaling nearly 1-1/2 million foot-years resulted in only one failure which was attributed to seven years of exposure to a concentrated caustic chemical powder. This 1951 cable had an asbestos fabric jacket instead of the neoprene type used since 1952.
2. To insure that the cables would not be damaged during the pulling operation, the maximum required pulling tension was calculated in 1970. These calculations showed that the maximum required tension would be less than one-third of the maximum allowable tension. Kerite engineers reviewed and concurred with these calculations. Calculations also indicated that the use of an approved pulling compound would reduce the maximum required pulling tension to less than one-sixth of the maximum allowable. Dynamometer checks during the actual cable pull measured the pulling tension at approximately one-seventh of the maximum allowable.
3. During cable installation visual spot checks were made to ascertain whether any damage was done during the cable pull. Cable was inspected after it was pulled into the Intake Structure pull box. No damage of any kind was detected.
4. To increase cable reliability, no cable splices were allowed in any of the cable runs.
5. The extensive experience which the Kerite Company and Baltimore Gas and Electric Company have had with this type cable indicates that there will be no deleterious effects due to aging during the life of the power plant.
6. The Kerite Company recommends the megger test for locating trouble without causing additional cable damage; also, the megohm readings will indicate trends toward insulation deterioration. In view of this recommendation, these saltwater pump motor feeders will be tested annually by a 2,500 Volt megger as a means of detecting any cable degradation.

TABLE 9-17A

SINGLE FAILURE ANALYSIS

DESCRIPTION OF LEVEL SWITCHES USED IN TABLE 9-17A

INTAKE STRUCTURE

Four level switches per unit are used. The Unit 1 side of the Intake Structure is separated from the Unit 2 side by a wall three feet high. Level switches on the Unit 1 side trip the Unit 1 circulating water pumps, and level switches on the Unit 2 side trip the Unit 2 circulating water pumps. Level switches are located at a height of 3" above floor, are Seismic Category I, are manufactured to special quality control requirements, are waterproof, and are testable. Level switches feed a two-out-of-four logic system located in the service building, which is testable when the plant is shut down. The logic system provides two outputs, either of which will trip all of the affected unit's individual pump circuit breakers. The system provides redundancy but not separation of components for tripping the pumps. The system also actuates an alarm in the Control Room for each unit.

CONDENSER PIT

Two level switches per location are used. Level switches are waterproof, testable, and are designed to function under seismic acceleration. The level switches feed a one-out-of-two logic system located in the Cable Spreading Room. The logic system is Seismic Category I, is testable, and actuates an alarm in the Control Room.

COMPONENT COOLING ROOM, ECCS ROOM AND SERVICE WATER ROOM

Two level switches per location are used. Level switches are waterproof, testable, and Seismic Category I. The level switches feed a one-out-of-two logic system located in the Cable Spreading Room. The logic system is Seismic Category I, is testable and actuates an alarm in the Control Room.

9.6 <u>SAMPLING SYSTEMS</u>

9.6.1 DESIGN BASIS

The sampling systems are designed to permit the sampling of liquids, steam, and gases for radioactive and chemical control of the plant primary and secondary fluids.

9.6.2 SYSTEM DESCRIPTION

The sampling system consists of six subsystems; reactor coolant sampling, steam generator blowdown sampling, radioactive miscellaneous waste sampling, turbine plant sampling, gas analyzing sampling, and post-accident sampling systems (PASS). Figure 9-10 shows the reactor coolant, the steam generator blowdown, PASS and the waste process sample systems. Figure 9-30 shows the turbine plant sample system. Figure 9-11 shows the gas analyzing system.

9.6.2.1 Reactor Coolant Sampling

Each reactor coolant sampling system consists of one stainless steel sink enclosed inside a hood. The hood is ventilated by an individual blower through a high-efficiency filter and located inside the sample room (Auxiliary Building). Interlocking high-density concrete block shielding separates the hood from the rest of the sample room, which also contains the steam generator blowdown system. The reactor coolant hood is used to determine the chemical and radiochemical condition of the reactor coolant and related auxiliary systems. The hood contains piping, valves, coolers, instrumentation, and sample bombs necessary to take liquid and gaseous samples from various systems. Two samples from the pressurizer (liquid, vapor) and one from the reactor coolant hot leg system can be controlled by three handswitches located on the steam generator blowdown panel. Should any one of the remotely-operated sampling valves fail to close after a sample is taken, a second remotely-operated valve can be shut from the Control Room. These valves are also closed by SIAS. The remotely-operated valves are backed up by manually-operated valves at the reactor coolant sampling hood. High-pressure samples flow through metering valves in order to reduce their pressure. One high-temperature sample is cooled in a sample cooler supplied with CC. All analyses on these samples are performed in the laboratory located in the Auxiliary Building.

9.6.2.2 Post-Accident Sampling

If needed (see Section 1.8.1, Item II.B.3), post-accident samples can be obtained in Unit 1 or Unit 2 Nuclear Steam Supply System Sample Room in the 45' Auxiliary Building. The Unit 1 or Unit 2 Nuclear Steam Supply System Sample Room contains piping, valves, coolers, and instrumentation necessary to sample either Unit 1 or Unit 2 RCS via either the normal RCS sampling line, or Unit 1 or Unit 2 Containment sump via the LPSI system header. A grab sample is used to obtain a liquid sample from the RCS. In the Chemistry Lab, the grab sample is depressurized, degassed, and diluted as necessary to enable handling the sample without excessive radiation exposure. This grab sample capability can be used to obtain samples from the RCS or the SI system. Sample purge waste is sent to the Reactor Coolant Drain Tank of the Unit being sampled, or alternatively to the Unit 1 or Unit 2 VCT. There is a provision to analyze the dissolved gasses in the liquid sample as well as chloride and boron. The gases from the degassed coolant are vented to atmosphere via Unit 2 Plant Vent via the Chemistry Lab hoods.

9.6.2.3 Steam Generator Blowdown Sampling

Each steam generator blowdown sampling system consists of one conditioning rack-panel unit and one ventilating hood, and is located inside the same sample room as the reactor coolant hood.

The conditioning rack section of the steam generator blowdown system contains isolation valves, primary coolers, rod-in-tube devices, an isothermal bath and chiller. High pressure samples are passed through a pressure-reducing valve (rod-in-tube type) located downstream of the primary coolers and upstream of the isothermal bath. High temperature samples first pass through a primary cooler (supplied with CC) and then through the isothermal bath. All samples pass through the isothermal bath which is capable of maintaining each sample at 77°F at the coil outlet. The chiller is supplied with cooling water from the component cooling system. Sample outlets from the conditioning rack are connected to the hood.

The panel section of the steam generator blowdown system contains conductivity and pH monitors, three hand switches for pressurizer sample selection, chiller controls, and an annunciator. The pH and conductivity samples are continuously monitored and alarmed on high conductivity. In addition, pH and conductivity are trended on the computer-based display in the chemistry laboratory. High sample temperature (downstream of the isothermal bath) actuates a common alarm point. Any point alarming on the local annunciator will actuate a master alarm in the Control Room (trouble alarm).

The ventilating hood contains two stainless steel sinks and is ventilated by an individual blower through a high-efficiency filter. The ventilating hood is used to obtain samples for determining the chemical and radiochemical content of the steam generator blowdown system. The radioactive miscellaneous sample system is also located inside the steam generator blowdown hood for Unit 1. The steam generator blowdown part of the hood contains all piping, grab sample valves, instrumentation including pH cells and conductivity analyzers, and all equipment necessary for this system.

9.6.2.4 Radioactive Miscellaneous Waste Sampling

The radioactive miscellaneous waste sampling is located inside the ventilating hood for the steam generator blowdown (Unit 1) and is used to obtain samples from which the chemical and radiochemical content of miscellaneous waste is determined. This system is common to both units. All samples are low pressure and are cooled, as necessary, in sample coolers (supplied with CC). This part of the hood contains isolation valves, piping, valves, and instrumentation necessary for obtaining liquid samples from both units. The analyses of these samples are performed in the laboratory located in the Auxiliary Building.

9.6.2.5 Turbine Plant Sampling System

Each turbine plant sampling system is used to obtain samples for determining the chemical condition of the steam, feed, and condensate systems associated with the turbine plant. The system consists of one sampling station per unit (stainless steel sink and panel) and one mechanical chiller as a separate unit. These sampling systems are located in the Turbine Building. The sink contains the isolation valves, piping, instrumentation, coolers, and grab valves necessary to take samples from the steam, condensate, and feedwater systems. High-pressure samples pass through a pressure-reducing valve (rod-in-tube device). All samples pass through individual primary coolers supplied with SRW. Every sample then

passes through cooling coils immersed in the isothermal bath that maintains each sample at 77°F at the coil outlet. The mechanical chiller circulates chilled water in the isothermal bath and is supplied with SRW. Each sample is provided with one grab sample valve for taking liquid samples as necessary. The steam generator feed pump headers are continuously monitored and recorded for hydrazine, oxygen and pH, any of which can cause an alarm on the annunciator. All samples are continuously monitored for conductivity and an alarm occurs when an abnormal condition is reached. In addition, samples are trended on the computer-based display in the Chemistry Laboratory. The turbine plant system contains conductivity, pH, and oxygen recorders, oxygen analyzers, handswitches (to control the hotwell sample pumps and the chiller circulating pump), and an annunciator. The annunciator alarms on high conductivity, high pH, high oxygen, high hotwell temperature, and low hotwell sample pump discharge pressures. Any annunciator alarm will activate a master alarm in the Control Room.

9.6.2.6 Gas Analyzing System

Control of hydrogen in Containment during and following a Design Basis Event is no longer required. On March 2, 2004, the NRC issued a license amendment that allows removal of the hydrogen recombiners and hydrogen analyzers from the Technical Specifications. The NRC has required retention of the hydrogen analyzers as non-safety-related equipment for recording hydrogen concentrations in a beyond Design Basis Event.

The gas analyzing system is used to determine the hydrogen concentration of six points inside the containment and of four samples from the reactor coolant waste tanks (receiver and monitor tanks), as well as the oxygen concentration of several samples from the reactor coolant and miscellaneous waste systems. The gas analyzing system is installed in the sample room located in the Auxiliary Building (Elevation-10') and consists of two hydrogen analyzer cabinets and separate manifolds for the isolation valves and sample selection solenoid valves and one oxygen analyzer cabinet with a manifold for the isolation valves. Two of the analyzer cabinets are for hydrogen measurement and include a sample pump, cooler, piping, valves, and instrumentation. Each hydrogen cabinet panel contains one hydrogen analyzer, one multipoint recorder for recording each measured sample, one programmer for random selection of individual readout, and alarm contacts for activation of a master alarm in the Control Room. The third analyzer cabinet is for oxygen grab sample measurement and includes a sample pump, cooler, piping, valves and sample syringe. An exhaust system on the oxygen analyzer cabinet purges any hydrogen that may leak into this cabinet.

The H_2 and 0_2 sample points are routed to the analyzer in accordance with the following table:

Sample Point	H_2 Analyzer 0-AE-6519	H_2 Analyzer 0-AE-6527	0_2 Grab Sample
Containment 1 - North of Primary Shield	No	Yes	No
Containment 1 - South of Primary Shield	Yes	No	No
Containment 1 – Pressurizer Compartment	Yes	No	No
Containment 1 - East at Elevation 135’	Yes	No	No
Containment 1 - West at Elevation 135’	No	Yes	No
Containment 1 - Dome at Elevation 189’[(b)]	No	Yes	No
Containment 2 - N	Yes	No	No
Containment 2 - S	No	Yes	No

Sample Point	H_2 Analyzer 0-AE-6519	H_2 Analyzer 0-AE-6527	0_2 Grab Sample
Containment 2 - Press	No	Yes	No
Containment 2 - E	No	Yes	No
Containment 2 - W	Yes	No	No
Containment 2 - Dome	Yes	No	No
RC Waste Rec Tank 11[a]	Yes	Yes	No
RC Waste Rec Tank 12[a]	Yes	Yes	No
RC Waste Mon Tank 11[a]	Yes	Yes	No
RC Waste Mon Tank 12[a]	Yes	Yes	No
Waste Gas Decay Tank 11	No	No	Yes
Waste Gas Decay Tank 12	No	No	Yes
Waste Gas Decay Tank 13	No	No	Yes
Waste Gas Surge Tank	No	No	Yes
Degasifier Accumulator 11	No	No	Yes
Degasifier Accumulator 21	No	No	Yes
Evaporators Discharge Gas Cooler	No	No	Yes
Przr. Quench Tank 11	No	No	Yes
Przr. Quench Tank 21	No	No	Yes
Misc. Waste Evap (Retired in place)	No	No	Yes

(a) These samples would normally be routed to either analyzer.

(b) The 189’ sample line in Unit 1 is inoperable because it is no longer seismically supported. For this reason it is not credited for post-accident sampling.

The six containment samples of the hydrogen analyzer cabinets and two samples of the oxygen analyzer cabinet can be controlled through remotely-operated solenoid valves.

To provide a post-accident containment air sampling capability, a sample vessel was placed into the sampling lines from containment 1 and 2 west at Elevation 135' to allow a syringe sample to be taken and analyzed in the laboratory. This sample vessel is located on the 45' Elevation of the Auxiliary Building.

9.6.3 SYSTEM RELIABILITY

All piping, tubing, fitting, and valves (exception listed under d and e below) in contact with fluids is 316 stainless steel and complies with the following codes:

a. ASME B&PV Code, Section III, Class 3 (Nuclear Power Plant Components) for the gas analyzing system.

b. ANSI B31.1 for turbine plant, steam generator blowdown, post-accident sampling, reactor coolant, and miscellaneous waste sampling systems. The exception to this is the normally-closed isolation valves located in the cabinets which constitute the boundary from ASME Section III piping to non-Class piping. These valves are listed below. (**NOTE**: The reactor coolant, the miscellaneous waste, and steam generator blowdown sampling systems, originally designed to meet Seismic Category I requirements, were downgraded to Category II via FCR 88-0074).

c. ASTM 450-68 which requires an eddy-current test for all tubing.

d. Pressure relief valves that are in contact with fluid shall be made of 304 or 316 stainless steel material.

e. Pressure reducing valves for the turbine plant and steam generator blowdown sample systems shall be constructed of Types 303, 304, or 316 stainless steel.

All applicable valves, piping, and coolers are designed to accept full steam pressure and temperature.

The gas analyzing system, the component cooling portion of the sample coolers in the reactor coolant hood, and the valves listed below are designed to meet Seismic Category I requirements. The reactor coolant, miscellaneous waste, and steam generator blowdown sampling systems (within the hoods and excluding those portions delineated above) are designed to meet Seismic Category II requirements. (**NOTE**: The reactor coolant, the miscellaneous waste, and the steam generator blowdown sampling systems, originally designed to meet Seismic Category I requirements, were downgraded to Category II via FCR 88-0074).

The following valves must be normally <u>closed</u> and will retain their current ASME Section III Seismic Category I classifications.

<u>Post-Accident Sampling</u>

1-PS-172	2-PS-172
1-PS-193	2-PS-193

<u>Miscellaneous Waste</u>

0-PS-226
0-PS-229

The following valves were Seismic Category I and designed in accordance with ANSI B31.1.

<u>Steam Generator Blowdown</u>

1-PS-126	2-PS-126
1-PS-128	2-PS-128
1-PS-129	2-PS-129
1-PS-137	2-PS-137
1-PS-139	2-PS-139
1-PS-140	2-PS-140

The turbine plant (Turbine Building) is designed to meet Seismic Category II requirements.

9.6.4 TESTING AND INSPECTION

Each component is inspected and cleaned prior to installation into the system. Instruments were calibrated during testing. Automatic controls were tested for actuation at the proper setpoints. Alarm functions were checked for operability and limits during preoperational testing period. The system will be operated and tested for flow, capacity, and mechanical operability.

9.7 FUEL AND REACTOR COMPONENT HANDLING EQUIPMENT

9.7.1 NEW FUEL STORAGE

New fuel is removed from its shipping container by the auxiliary hook of the Spent Fuel Cask Handling Crane and is transferred to the new fuel storage racks. These dry storage racks are for both units and are constructed to provide storage for two-thirds of a core (144 assemblies). New fuel, with a maximum enrichment of 5.0 wt% U-235, may be stored in the new fuel storage racks. For the Westinghouse standard fuel design, this results in a maximum effective multiplication factor of 0.89 at a water density of 1.0 gm/cc (full flood), and a multiplication factor of less than 0.89 for aqueous foam. Due to the large available margin, an uncertainty analysis was not performed since typical uncertainty analyses result in uncertainties of less than 3.0%. For the Westinghouse value added pellet (VAP) fuel design and AREVA/Framatome fuel assemblies, this results in a maximum effective multiplication factor of less than 0.95, including all biases and uncertainties for full flood and aqueous foam conditions. If there is space in the SFP, new fuel may be stored in the Unit 1 SFP provided its wt% U-235 does not exceed the maximum enrichment allowed in the Unit 1 SFP. New fuel may be stored in the Unit 2 SFP provided that the enrichment-burnup and checkerboarding restrictions of Limiting Condition for Operation 3.7.17 are met.

Unless specified, the reactivity of any SFP or refueling pool system completely filled with VAP assemblies with axial blankets and with or without a Zirc Diboride (ZrB_2) coating is always less reactive than the design-basis VAP configuration. Thus, VAP assemblies with enrichment up to 5.0 w/o, with axial blankets, and with or without ZrB_2 can be safety stored in the new fuel storage racks.

9.7.2 SPENT FUEL STORAGE

9.7.2.1 Spent Fuel Pool Racks

The SFP is located outside the containment in the Auxiliary Building and provides underwater storage of spent fuel assemblies after their removal from the reactor vessel. The pool, designed in two halves, can accommodate 1830 assemblies and one spent fuel shipping cask. The Unit 1 half of the SFP contains storage racks in six 10x10, two 8x10, and one 7x10 array. The Unit 2 half of the SFP contains racks in ten 10x10 arrays. Control element assemblies removed from the core can be stored in the guide tubes within the fuel assemblies. The pool is constructed of reinforced concrete and lined with stainless steel. The pool was designed in accordance with Safety Guide No. 13, published March 10, 1971.

The spent fuel assemblies are placed in stainless steel storage racks consisting of vertical cells grouped in parallel rows with a center-to-center distance of 10-3/32" in both Units. Sandwiched between the inner and outer walls of each storage cell is a 6.5" wide sheet of B_4C poison material. Unit 1 storage racks use a B_4C composite material, carborundum, and Unit 2 racks use Boraflex. There is a coupon surveillance program to monitor the condition of the carborundum material. Boraflex is no longer credited as a neutron absorber due to degradation calculations (License Amendment No. 246); therefore, testing the Boraflex coupons is no longer necessary and has been eliminated. The top opening of the racks has angled lead-in guides which effectively block the spaces between the cavities, as well as guide the fuel assembly into the open tube.

Per Title 10 Code of Federal Regulations (CFR) 50.68, if no credit for soluble boron is taken, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with unborated water. If credit it taken for soluble boron, the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel

assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical) at a 95% probability, 95% confidence level, if flooded with unborated water. In addition, the maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to 5 wt%.

For an infinite axial and radial array of Unit 1 storage cells of nominal dimensions with credit for the carborundum poison sheets and containing the maximum enrichment of 5.0 w/o Westinghouse VAP fuel at the worst case temperature of 40°F, the maximum unborated k_{eff} value of 0.986 is calculated with all biases and uncertainties, which is less than the 10 CFR 50.68 regulatory value of 1.0. The maximum k_{eff} value of 0.947 at a moderator boron concentration of 350 ppm with all biases and uncertainties is less than the 10 CFR 50.68 regulatory value of 0.95. Note that Westinghouse VAP fuel is more reactive than similarly enriched Westinghouse standard fuel and AREVA fuel, thus any analysis performed for Westinghouse VAP fuel conservatively bounds that for Westinghouse standard fuel and AREVA/Framatome fuel.

Possible boraflex degradation in the Unit 2 SFP was documented based on calculations using the Racklife software package. Crediting burnup in lieu of boraflex assures that the 10 CFR 50.68 regulatory k_{eff} limit is maintained. The CCNPP SFP Rack Criticality Methodology ensures that the spent fuel rack multiplication factor, k_{eff}, is less than the 10 CFR 50.68 regulatory limit with Westinghouse VAP and standard fuel, and AREVA/Framatome fuel ranging in enrichment from 2.0 to 5.0 w/o with burnup credit and with partial credit for soluble boron in the Unit 2 SFP. The soluble boron credit will be limited to 350 ppm per the restrictions of the Unit 1 criticality analysis including all biases and uncertainties.

The burnups required to store fuel in the Unit 2 SFP crediting 350 pm of soluble boron including all biases and uncertainties are the following:

Enrichment (w/o)	Burnup (GWD/MTU)
2.0	6.00
2.5	13.75
3.0	20.50
3.5	27.00
4.0	32.75
4.5	38.25
5.0	43.75

Each assembly offloaded from either reactor or from an Independent Spent Fuel Storage Installation dry shielded canister must be evaluated against the above burnup restrictions to determine if it can be safely stored in the Unit 2 SFP. No similar restrictions exist on the Unit 1 SFP.

Several checkboard patterns were modeled in an effort to store more reactive fuel in the Unit 2 SFP. Note that only one pattern meets the requirements of 10 CFR 50.68. If credit is taken for soluble boron, that fuel assembly must be surrounded on all four adjacent faces by empty rack cells or other nonreactive materials (e.g., wall, water, …).

The SCALE 4.4 CSAS25 code module with the 44 group ENDF/B-V cross-section library was utilized to perform the KENO-Va Monte Carlo criticality calculations. The neutron adsorption of the stainless steel rack cells was credited in the criticality calculations. However, no credit for the U-234 and U-236 fuel was taken.

The analysis methods and neutron cross-section data are benchmarked by comparison with critical experiment data for similar configurations. The benchmarking process establishes a calculational bias and uncertainty of the mean with a one-sided tolerance factor of 95% probability at a 95% confidence level. The maximum k_{eff} value for the SFP is obtained by summing the calculated value, the calculational bias, the total uncertainty defined as a statistical combination of the calculational and mechanical uncertainties, and the burnup axial distribution bias. Mechanical and material uncertainties may be treated by assuming worst case conditions or by performing sensitivity studies and obtaining worst case uncertainties. Uncertainties may be combined statistically provided that they are independent variations.

The fuel design uncertainty analysis of the Unit 1 SFP consists of the following:

	Westinghouse	AREVA/ Framatome
Delta k_{eff} for 95/95 calculational uncertainty	0.00760	0.00760
Delta k_{eff} for stack height density	0.00417	0.00267
Delta k_{eff} for storage cell pitch	0.00575	0.00571
Delta k_{eff} for steel thickness	0.00569	0.00462
Delta k_{eff} for poison loading	0.00607	0.00513
Delta k_{eff} for eccentric positioning	0.00249	0.00243
Delta k_{eff} for fuel pellet diameter	0.00000	0.00148
Delta k_{eff} for water in gap	0.00000	0.00574
Total Delta k_{eff}	0.01355	0.01365

The Unit 1 SFP Biases included:

Delta k_{eff} for calculational methodology	0.00080	0.00080
Delta k_{eff} for poison loading	0.00466	0.00000
Total Delta k_{eff}	0.00546	0.00080

The Unit 1 SFP worst-case assumptions are the following:

Temperature	4°C (39.2°F)	4°C (39.2°F)
Fuel Clad Composition	Optin	M5®
Fuel Enrichment	5 wt%	5 wt%

A total Unit 1 bias and uncertainty value of 0.01901 was included in all Unit 1 SFP reactivity results for Westinghouse fuel, and a value of 0.01445 was included in all Unit 1 SFP reactivity results for AREVA/Framatome fuel.

The fuel design uncertainty analysis of the Unit 2 SFP consists of the following:

	Westinghouse	AREVA/ Framatome
Delta k_{eff} for 95/95 calculational uncertainty	0.00760	0.00760
Delta k_{eff} for stack height density	0.00090	0.00394
Delta k_{eff} for storage cell pitch	0.00358	0.00711
Delta k_{eff} for fuel enrichment	0.00155	0.00696
Delta k_{eff} for steel thickness	0.01346	0.01353
Delta k_{eff} for eccentric positioning	0.00961	0.01380
Delta k_{eff} for fuel depletion	0.02089	0.01956
Delta k_{eff} for fuel pellet diameter	0.00000	0.00185
Delta k_{eff} for water in gap	0.00000	0.00269
Delta k_{eff} for clad composition	0.00000	0.00252
Total Delta k_{eff}	0.02799	0.03075

The Unit 2 SFP Biases included:

Delta k_{eff} for calculational methodology	0.00080	0.00080
Delta k_{eff} for axial burnup distribution	0.03250	0.03030
Total Delta k_{eff}	0.03330	0.03110

The Unit 2 SFP worst-case assumptions are the following:

Temperature	68.33°C (155°F)	68°C
Fuel Clad Composition	Zirc4	M5®
Poison Loading	0.000 gm/cm^2	0.000 gm/cm^2

A total Unit 2 bias and uncertainty value of 0.06129 for Westinghouse fuel and 0.06185 for AREVA/Framatomre fuel was included in all Unit 2 SFP reactivity results, where burnup credit was assumed. If no burnup or burnup credit is assumed, a total Unit 2 bias and uncertainty value of 0.01944 for Westinghouse fuel and 0.02452 for AREVA/Framatome fuel may be assumed.

A finite radial and axial model of the Unit 1 SFP of nominal dimensions containing the maximum enrichment of 5.0 w/o fuel at the worst case temperature of 40°F at a soluble boron concentration of 350 ppm including all biases and uncertainties was modeled with alternate and sequential assemblies in the row closest to the SFP wall on spacers to simulate the reconstitution/inspection process. Sufficient margin exists in going from a two to three dimensional model to counteract any increase in reactivity from raising a row of assemblies on spacers. In addition, there is no reactivity penalty between reconstituting an entire row of assemblies or alternate assemblies in a row. Since the boraflex is no longer credited in the Unit 2 SFP, raising assemblies on spacers has no reactivity effect.

Dropping an assembly of 5.0 w/o fuel onto the SFP racks was analyzed, even though it is not a credible accident. The double contingency principle was applied, which required two unlikely, independent, concurrent events to produce a criticality accident. The double contingency principle means that realistic conditions may be assumed. For example, if soluble boron is normally present in the SFP water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, total credit for the presence of soluble boron may be assumed in evaluating this accident condition. Per Technical Specifications, the normal SFP boron concentration is conservatively assumed to be 2000 ppm. Taking credit for 2000 ppm per the double contingency principle drops the k_{eff} value for this accident to well below the regulatory requirement.

The racks are designed to withstand all anticipated loadings. Structural deformations are limited to preclude any possibility of criticality. The racks are supported in such a manner as to preclude a reduction in separation space under either the Operating Basis or Safe Shutdown Earthquake. The racks themselves are designed not to collapse or bow under the force of a fuel assembly dropped into an empty cavity, or dropped horizontally across the top of the racks assuming no drag resistance from the water. The structure is fabricated of stainless steel and boron carbide sheets in both units and meets the requirements of Seismic Category I.

Spent fuel decay heat is removed by the fuel pool cooling system described in Section 9.4. The design of the racks allows for adequate convective cooling of stored fuel assemblies by natural circulation.

Monitoring and alarm instrumentation are provided at appropriate locations to assure that the decay heat from the spent fuel elements is being removed and to assure that proper radiation levels are maintained. Means will be provided to control unauthorized entry and to account for the flow of tools in and out of the area.

9.7.2.2 ICI Trash Container Rack

A four-cell ICI trash container rack is located in the lower portion of the refueling canal adjacent to the upender machine. The rack is positioned such that assemblies or cans are submerged to the same elevation as when vertically in place in the upender. Incore instrumentation trash containers or new/spent fuel assemblies may be temporarily placed in the rack to facilitate handling during refueling. The reactor vessel closure head guide studs are stored in the rack during normal plant operation. The guide studs align the reactor vessel head during refueling outages and do not interfere with the temporary handling of fuel assemblies or the ICI trash containers. The rack is stainless steel with four vertical storage positions on 24" centers. The ICI storage rack is designed to withstand all anticipated loadings and meets the requirements of Seismic Category I. Open frame construction allows for natural convective cooling.

The ICI rack design includes members which are located on the tops and sides of the racks to prevent both the inadvertent insertion of an assembly between already stored assemblies and the transportation of an assembly to a position directly adjacent to already stored assemblies. Sufficient distance is provided between the top of the active fuel and the top deck of the storage rack to preclude criticality in the event that a fuel assembly is dropped and lands in a horizontal position on top of active fuel. Angled lead-ins guide an assembly or trash can into the rack and prevent inadvertent insertion of an assembly between already stored assemblies. The criticality analysis supports the Westinghouse standard and VAP design and the AREVA/Framatome design for 5.0 wt% U-235 enrichment for the ICI storage racks. Four fresh 5.0 wt% assemblies stored in the ICI racks with a fifth positioned at the minimum standoff distance, assuming no soluble boron, will maintain k_{eff} less than 0.95 including all biases and uncertainties. The k_{eff} will also remain less than 0.95, including all biases and uncertainties, after dropping a fresh 5.0 wt% assembly on an ICI storage rack filled with four fresh 5.0 wt% assemblies, assuming no soluble boron.

9.7.2.3 Spent Fuel Shipping Cask Pit

The spent fuel shipping cask pit is located on the Unit 1 side of the dividing wall in the pool. The floor of the pit is equipped with a stainless steel cask support platform upon which a shipping cask is set before being loaded with spent fuel bundles. Every cask used is designed such that spent fuel bundles can be placed in them while still maintaining the minimum water level above the fuel bundles. The cask cover is then placed on the cask and the unit is transferred to the cask wash down area by the Spent Fuel Cask Handling Crane. The wash down water is then piped to the MWPS. Means will be taken to assure that surface contamination is less than required by transportation regulations

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A cask platform/energy absorbing device is located in the cask pit area and provides two functions:

a. Elevates the pit floor surface so that the lifting trunnions on the NUHOMS transfer cask do not interfere with the cask seismic restraint.

b. Provides a second level of protection beyond that provided by the single-failure-proof crane in that the platform has the capability of absorbing the energy associated with a crane drive train failure.

The energy absorbing cask support platform, located inside the cask pit area, is comprised of a stainless steel shell Hexcel aluminum honeycomb material designed to meet the requirements of Seismic Category I and to protect the floor of the cask pit area by absorbing the impact of a cask due to drive train failure of the Spent Fuel Cask Handling Crane.

9.7.2.4 Spent Fuel Cask Handling Crane

Heavy loads (loads in excess of 1600 lbs) are prohibited from travel over spent fuel assemblies in the SFP unless such loads are handled by a single-failure proof device. The Spent Fuel Cask Handling Crane, which is designed in accordance with the "single-failure-proof" criteria of NUREG-0554 and NUREG-0612, is used to handle heavy loads in the SFP area. The maximum design rated load for the Spent Fuel Cask Handling Crane is 150/15 ton (150 ton for the main hoist and 15 ton for the auxiliary hoist). Its maximum critical load rating is 125/15 ton.

The Spent Fuel Cask Handling Crane is used to handle casks over the spent fuel pool and surrounding structures. The crane is single-failure-proof and has been designed in accordance with NUREG 0554.

9.7.2.5 SFP Purification

The SFP purification system consists of a demineralizer and filter. The demineralizer is not regenerated. When the demineralizer or filter is depleted, or as necessary, they are replaced. Additionally, cask movement meets all criteria of NUREG-0612.

The height of the filter transfer cask is 5'6" (including lifting rig). The monorail hoist hook Elevation is 57'0", 12' above the floor. Therefore, there will be adequate clearance between a cask and the floor.

The filter transfer cask and the SFP purification filter together weigh 6.30 tons. The monorail and hoist is rated for 7.5 tons.

9.7.2.6 SFP Ventilation

The spent fuel handling area ventilation system, shown in Figure 9-21, contains charcoal filters, which remove iodine and other radioactive particulates. The Auxiliary Building air is discharged to the plant vent which is constantly monitored.

9.7.2.7 Spent Fuel Handling Machine and New Fuel Elevator

The spent fuel handling machine is located above the SFP. It is a bridge and trolley arrangement, similar to the refueling machine, which rides on rails set in the concrete on each side of the pool. The handling machine is designed to pass over the dividing wall (separating the two halves of the pool) and to serve both halves of the pool. Latitude and longitude motors on the bridge and trolley position the machine over the specified rack location in the SFP
.

The spent fuel handling machine serves several purposes, some of which are given here. One purpose is to transfer the spent fuel from the upending mechanism to a location in the SFP for decay, or to transfer new fuel to the upending machine. A second purpose is to take fuel from the new elevators and transfer it to a rack location in the SFP or to the fuel upending mechanism. A third function is to move fuel to and from the spent fuel inspection elevator, inspection/repair stations, and to move fuel between storage rack locations. A fourth function of the spent fuel handling machine is to transfer the decayed spent fuel to the shipping cask.

The spent fuel handling machine and new fuel elevators are designed to Seismic Category I requirements. The new fuel elevator is mounted on the west side of the Unit 1 SFP. The function of this elevator is to transport new fuel assemblies with a maximum enrichment of 5.0 wt% U-235 in the pool where the spent fuel handling machine is able to grapple and transfer the fuel to the desired location in the SFP. The fuel elevator on the west side of the Unit 2 SFP was modified to allow its use in inspection of fuel assemblies with a maximum enrichment of 5.0 wt% U-235. Two standoffs located on the new fuel elevator box assemblies ensure that a fuel assembly suspended from the spent fuel handling machine cannot be brought within eight inches of new fuel in the new fuel elevators. This is an added measure, along with existing interlocks and administrative controls, for the prevention of criticality. The spent fuel handling machine and fuel elevators are shown in Figures 1-9 and 1-13.

The Spent Fuel Handling Machine (SFHM) is capable of four modes of operation:

a. *Manual* mode allows movement of SFHM bridge, trolley, hoist, and grapple without system power available to the SFHM.
b. *Manual-Electric* mode allows SFHM bridge and/or trolley movements via joystick operations.
c. *Semi-Automatic* mode allows the operator to set "from" and "to" locations via the console, and the SFHM will automatically move the bridge and/or trolley per those settings.
d. *Automatic* mode allows SFHM bridge and/or trolley movements per a pre-determined file that contains a range of "from" and "to" locations.

Some of the safety features incorporated in this equipment are interlocks to prevent movement into the walls. These interlocks can be bypassed and restored in accordance with approved procedures. Additional safety features include limit switches to prevent the hoist from raising fuel above the point where adequate water for shielding is available. A redundant mechanical stop will prevent a fuel bundle from being raised above specified limits. This results in a maximum dose rate of 7 mrem/hr over the pool during refueling operations. The fuel grappling tool is designed so that the fuel bundle cannot be released accidentally. All motors are equipped with mechanical brakes or self-locking gears to prevent movement in case of a loss of power.

9.7.2.8 Spent Fuel Pool Platform

The SFP platform is a 16' long, 4' wide platform that fastens to the side of the SFP. It is designed such that when installed it will not interfere with the operation of the fuel handling machine. Removable railings are provided for personnel safety. The work platform is portable and can be located along the west wall of the north pool or the east wall of the south pool.

The original purpose of the work platform was to provide an efficient work site for the repairing of worn fuel assembly guide tubes (Section 3.6). The work platform overhang allows repairs to be made from a position directly over the fuel assemblies. The platform was first installed in the south pool in September 1978 in preparation for Unit 2's first refueling. It was moved to the north pool in March 1979 for Unit 1's third refueling. Since then the platform has been moved between the two pools as needed. Subsequently, its use has expanded to include eddy current tests, capsule exchanges, and fuel assembly reconstitutions.

9.7.2.9 Independent Spent Fuel Storage Installation

A detailed description of the Independent Spent Fuel Storage Installation and the transfer operations is discussed in the Independent Spent Fuel Storage Installation Safety Analysis Report.

9.7.2.10 Storage of Failed Fuel Rods in Encapsulation Tubes

Encapsulation tubes are a standard Asea Brown Boveri/Combustion Engineering device for storing failed fuel rods and for containing solid fission products. They are easily identifiable and retrievable. Encapsulation tubes safely store individual irradiated failed fuel rods in the SFP in the peripheral guide tubes of empty grid cages. A single encapsulation tube containing a damaged fuel rod can be stored in an ICI trash can, can be stored in an empty SFP rack space that is inaccessible to both the SFHM and the Auxiliary Building cask handling crane, can be laid temporarily atop the spent fuel pool storage racks with administrative restrictions on fuel movement in the laydown area, or can be placed at the bottom of an upender trench with the associated upender tagged out. Failed fuel rods in encapsulation tubes cannot be stored in the center guide tube of an empty grid cage, since an encapsulation tube from the center guide tube of a grid cage can become wedged in the grapple of the SFHM. Encapsulated fuel rods stored within the guide tubes of empty grid cages or stored in an ICI trash can or empty SFP rack space, are prohibited from extending above the spent fuel pool racks to avoid interfering with the SFHM and its load.

A criticality incident in the SFP will not occur. Storage of the encapsulation tubes in the peripheral guide tubes of empty grid cages or in an ICI trash can or empty SFP rack space will cause a decrease in maximum SFP reactivity due to a decrease in fissile inventory and will not create the possibility of inadvertent criticality. An encapsulated fuel rod placed temporarily atop the spent fuel pool storage racks or at the bottom of the upender trench will be decoupled in reactivity space from the assemblies stored within the rack. Undamaged fuel rods can only be stored in the encapsulation tubes in empty grid cages. This will ensure that the consequences of a fuel handling incident will be limited by the current analysis.

9.7.3 REACTOR COMPONENT HANDLING EQUIPMENT

The refueling equipment arrangement is shown in Figure 9-12.

9.7.3.1 Reactor Refueling Machine

The reactor refueling machine is shown in Figure 9-13.

The refueling machine is a traveling bridge and trolley which spans the refueling pool, and moves on rails located at Elevation 69'6" in the containment area. 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 bundle to another location or to the upender.

The controls for the refueling machine are mounted on a console which is 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 handwheels are provided for bridge, trolley and winch motions in the event of a power loss.

The Refueling Machine is capable of three modes of operation:

a. *Manual-Electric* modes allow Refueling Machine bridge and/or trolley movements via joystick operation.
b. *Semi-Automatic* mode allows the operator to set “from” and “to” locations via the console, and the Refueling Machine will automatically move the bridge and/or trolley per those settings.
c. *Automatic* mode allows Refueling Machine bridge and/or trolley movements per a pre-determined file that contains a range of “from” and “to” locations.

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. Variations from normal loads in excess of 10% will automatically stop the motion of the hoist winch mechanism. 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 mast assembly and is operated after grappling of the fuel assembly. 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 guide studs, structures within the refueling cavity or the walls of the refueling cavity;
b. Interlocks which restrict simultaneous operation of either the bridge and trolley or the hoist winch drive mechanism;
c. An interlock which prevents bridge and trolley motion when spreader device is actuated;
d. Interlocks that prevent bridge and trolley motion until the hoisting operation is complete;
e. Over and under load switches which stop fuel hoist motion;
f. Automatic bridge and trolley speed restriction zones over the reactor core;
g. Fuel hoist programmed speed restriction while the fuel bundle is within the core and upending machine;
h. An interlock which prevents positioning of the refueling machine over the tilting machine unless the hoist is at the up limit and the spreader is retracted.

9.7.3.2 Fuel Transfer System

Upending Machines

Two upending machines are provided for each unit, one in the Containment Structure refueling pool and the other in the SFP. 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 between a vertical and horizontal position, as required by the fuel transfer procedure.

Interlocks are provided to ensure the safe operation of this equipment by prohibiting the lowering of a fuel assembly unless the fuel carrier is vertical, by preventing inadvertent rotation of the tilting cylinders while a fuel assembly is being lowered, and by deactivating the cable drive so that a premature attempt to move the carriage through the transfer tube cannot be initiated.

Fuel assemblies of 5.0 wt% U-235 can be inserted into the upenders. A k_{eff} less than or equal to 0.95 with biases and uncertainties can be maintained with two fuel assemblies of 5.0 wt% U-235 inserted into the upender.

Transfer Carriage

During refueling periods, the transfer carriage transports one or two fuel assemblies with a maximum enrichment of 5.0 wt% U-235 between the refueling pool and the fuel storage area. Ten large wheels, five 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 mechanisms. The carriage is driven by steel cables connected to the carriage and through sheaves to its driving winches mounted below the operating floor level. The fuel carrier is mounted on the carriage and is pivoted for tilting by the upending machines.

Transfer Tube and Isolation Valve

The fuel transfer tube shown on Figure 9-14 connects the refueling pool with the SFP. During reactor operation, the transfer tube is closed by an isolation valve outside the containment and a blind flange inside the containment (Figure 9-14A). The flange is subject to local leak rate testing (Type B). The isolation valve is not local leak rate tested. 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-type expansion joint.

Transfer Rails

This assembly contains the rails on which the transfer carriage rides when moving between the reactor cavity and SFP area. The rail supports are welded to the 36" diameter transfer tube. The rail assemblies are fabricated to a length which will allow 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.

9.7.3.3 CEA Handling Tool

Unit 1:

The refueling machine auxiliary hoist, used in conjunction with the CEA handling tool, is used to exchange CEAs within the reactor core under normal conditions. The auxiliary hoist has sufficient capacity to hoist a CEA.

Unit 2:

The refueling machine auxiliary hoists, used in conjunction with the CEA handling tool, are used to exchange CEAs within the reactor core under normal conditions. Each auxiliary hoist has sufficient capacity to hoist a CEA.

The CEA handling tool is visually aligned by a licensed operator to verify that the tool is correctly positioned above the fuel assembly prior to grappling. The CEA handling tool has a rotary grapple which rides in a vertical channel section so that inadvertent release of a CEA is not possible. A load cell is used with the CEA handling tool to verify loading (unloading) and prevent hoisting a bound CEA. Administrative controls prevent translation during hoisting or vice versa.

The CEA handling tool can also be used in the SFP, where it is lifted by the single failure proof crane.

9.7.3.4 Reactor Vessel Head Lifting Rig

The reactor vessel head lifting rig is shown in Figure 9-16.

This lifting rig is composed of a three-part lifting frame (tripod) and three lift links. The lift links are attached to the outer shroud, which is part of the CEDM air cooling structure. The outer shroud is attached to the reactor vessel head. The lift links support a service structure that includes three hoists for handling the hydraulic stud tensioners, reactor vessel studs, washers, and nuts. The lift links and service structure provide support for the CEDM, Reed Switch Position Transmitter, ICI, and Reactor Vessel Level Monitoring System electrical cables. The tripod is removed prior to plant operation.

9.7.3.5 Reactor Internals Lifting Rig

The reactor internals lifting rig consists of three major subassemblies: (1) upper guide structure lift rig which includes an ICI hoist, (2) a core support barrel lifting rig, and (3) an upper clevis (tripod) assembly. The upper clevis assembly is common to the upper guide structure lifting rig and the core support barrel lifting rig. The upper clevis assembly is a tripod-shaped structure connecting the lifting rigs to the containment crane lifting hook.

The upper guide structure lifting rig is shown in Figure 9-17. This lifting rig consists of a delta spreader beam which supports three columns providing attachment points to the upper guide structure. Attachment to the upper guide structure is accomplished manually from the working platform by means of lifting bolt torque tools. The integral ICI hoist connects to an adapter which is manually attached to the ICI structure by utilizing an adapter torque tool. The ICI is then lifted by the crane hook.

A core support barrel lifting rig, shown in Figure 9-18, is provided to withdraw the core barrel from the vessel for inspection purposes. The lifting rig includes a spreader beam providing three attachment points. Attachment is accomplished manually from the refueling machine bridge by means of a lift bolt torque tool.

Correct positioning of either the upper guide structure lifting rig or the core support barrel lifting rig is assured by guide bushings attached to the rigs which mate to the reactor vessel guide pins.

A separate upper guide structure lift rig is provided for each of the two reactors. The core support barrel lifting rig is shared by Unit 1 and Unit 2.

9.7.3.6 Surveillance Capsule Retrieval Tool

A retrieval tool is provided during the refueling shutdown for manual removal of the irradiated capsule assemblies of the reactor vessel materials surveillance program described in Section 4.1.5.4.

A diagram of the surveillance capsule retrieval tool is shown in Figure 9-19. The tool is operated from a position on the carriage walkway of the refueling machine. Access to the capsule assembly is achieved by inserting the tool through 3" diameter retrieval holes in the core support barrel flange provided in each capsule assembly location. A female acme thread at the end of the retrieval tool is mated to the surveillance capsule lock assembly (Figure 4-14) by turning the retrieval tool handle. A compressed spring in the lock assembly exerts a high frictional force at the retrieval tool-lock assembly interface to prevent disengagement during retrieval.

The overall length of the tool is 45.5'. The tool consists of two parts to facilitate storage. The upper portion is a 2" diameter tube and handle. The lower portion of the tool is also a 2" tube with a 1" outer diameter at the connector end. A 3/4" diameter hole in the upper end of the tool permits the containment crane to assist with the retrieval procedure and prevents inadvertent dropping of the tool. The tool is made of aluminum and has a dry weight of 40 lbs.

9.7.4 DESIGN EVALUATION AND SYSTEM RELIABILITY

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 and recorded (Section 11.2.3). 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 during fuel loading and unloading operations.
b. Whenever fuel is added to the reactor core, the source range neutron flux (count rate) is recorded to verify the subcriticality of the core.
c. The design of the equipment places physical limits on the extent of fuel movement, thereby avoiding any 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 fresh water. The design of the spent fuel storage pool is such that water cannot drain out of the pool by gravity.

Manually-operated handwheels are provided to allow refueling bridge, trolley and winch 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 prior to each refueling operation.

At least 10" is provided between the top of the active fuel and the top deck of the storage rack to preclude criticality in the event of a fuel assembly is dropped and lands in the horizontal position on the top deck. The design of the racks assures adequate convective cooling to a fuel assembly lying horizontally across the top of the racks.

9.8 PLANT VENTILATING SYSTEMS

9.8.1 DESIGN BASIS

The plant ventilating systems are designed to provide a suitable environment for equipment and personnel with a maximum amount of safety and operating convenience. Potentially contaminated areas are separated from clean areas. Airflow patterns originate in areas of potentially low contamination and progress toward areas of higher activity. Generally, negative pressures are maintained in potentially contaminated areas and positive pressures in clean areas. The ventilating systems in the containment, waste processing and fuel-handling areas are designed for containment of radioactive particles. The path of the discharge from all potentially contaminated areas is directed into the respective plant vent where the radioactivity level is monitored. The equipment in most critical systems is redundant in character; detailed descriptions are presented where this occurs. Basic temperature design criteria are listed in Table 9-18.

9.8.2 SYSTEM DESCRIPTION AND OPERATION

9.8.2.1 General

The plant ventilation systems discussed in this section are shown on Figures 9-20A, 9-20B, and 9-21, and listed in Table 9-18.

The containment cooling and filtering systems are discussed in Sections 6.5 and 6.7, respectively. The penetration room ventilation system is discussed in Section 6.6.

9.8.2.2 Containment Ventilation

Control Element Assembly Drive Mechanism Cooling System

In this system, air is drawn from the containment at a rate of 800 cfm and design temperature of 120°F, through the reactor head cooling shroud and into two cooling coils of the CEDM cooler, which is located on the missile shield above the reactor. From there, 100% redundant fans discharge the cooled air upward into the containment again. Four ducts connect the shroud to the cooler coil house. One pair of ducts directs air to one cooling coil and the other pair supplies air to the opposite coil. Cooling water at a design inlet temperature of 95°F is pumped through the water-air coils. A power-operated damper located between each fan and the coil house prevents short-circuiting of air around the cooler when only one fan is operating. The switch-over from one fan to the other is accomplished by remote-manual control from the Control Room. Tests have shown this cooling air is more than adequate to maintain coil temperatures below 350°F. The airflow and geometrical cooling shroud configuration simulated that of the on-site installation. Testing of the CEDMs included holding, insertion, withdrawal and tripping operation.

In no case will loss of cooling air prevent the CEDM from releasing the CEAs if a reactor trip is initiated. Tests, in a simulated operating environment, have shown that the CEDM is capable of dropping the CEA after four hours of operation in the hold mode without cooling air supply. These heat transfer tests were conducted on a full-size prototype in a hot autoclave simulating reactor operation conditions.

Containment Purge System

There is a separate, identical purge system for each containment. In each system, an air-handling unit, located in the Auxiliary Building, supplies filtered and tempered air to the containment through a supply duct.

One exhaust fan for each Containment Structure, located in the Auxiliary Building, draws air from the containment through an exhaust duct and high efficiency particulate air (HEPA) filters, and discharges it into the respective main plant exhaust plenum where the fans force it into the plant vent. The air-operated butterfly valves, which fail closed, are located in the supply and exhaust ducts inside of containment to provide containment closure when the reactor is in a shutdown condition. During reactor operation, the purge penetrations are closed by a blind flange in each penetration outside of containment.

When the reactor is shut down, the containment purge isolation valves are closed and the purge system fans are stopped by a containment radiation signal.

During normal operations of the containment purge ventilation system, negative pressure is maintained in the Containment. Alternate line-ups may result in some natural air circulation in and out of Containment. Administrative procedures are in place to monitor and ensure that the potential release of radioactive particles from the Containment, while the purge air supply and exhaust fans are secured, will remain within the Offsite Dose Calculation Manual limits.

Containment Vent System

This system is designed to operate as a containment vent during power operations. The system is utilized to control containment pressure and airborne radioactivity within specified Technical Specification limits.

Upon receipt of a SIAS, containment radiation signal, or a high-radiation signal, the inboard and outboard MOVs close.

Although control of hydrogen in Containment following an accident is not required, this penetration may be used as a hydrogen purge.

Pressurizer Compartment Cooling

A metal wall, designed to blow out at less than 5 psi, separates the pressurizer compartment from the RCP area so as to prevent entrance of hot air from the pump motors into the compartment. In order to prevent the concrete upon which the pressurizer rests or pipe-mounted electrical components from overheating, cooling air is supplied at two levels from the containment coolers. In addition, air is supplied from the containment to the upper extremes of this compartment. The air supplied for cooling pressurizes this compartment and then exits through the access opening at Elevation 81'0".

Cavity Cooling System

Two redundant fans supply air from the containment air cooler plenum through ducting to the reactor cavity distribution manifold where it is used to cool the neutron detectors, the primary shield penetrations and the primary shield. System performance is adjusted by manual balancing. High efficiency filters are installed to protect each branch which serves a neutron detector.

9.8.2.3 Auxiliary Building Ventilating Systems

Control Room

The Control Room (Elevation 45'0") and the Cable Spreading Room (Elevation 27'0") are incorporated into a single year-round air-conditioning system serving both Units 1 and 2. Therefore, the ambient temperature in the Control Room is expected to be the same as the ambient temperature in the Cable Spreading

Room. Air handling and refrigeration equipment are redundant. The Control Room and Cable Spreading Room areas have a third source of cooling, which is not safety-related, in the form of a water chiller supplying a second set of coils in the safety-related air handling systems.

In the event that both the non-safety-related chiller and the safety-related condensers are rendered inoperable by a tornado, a post-tornado mode of cooling the Control Room and cable spreading rooms is available. In this mode of cooling, the fresh air dampers are fully opened, the recirculation dampers are fully closed, and the exhaust damper is fully opened to allow Control Room and cable spreading room cooling using outdoor air only.

The Control Room ventilation system continuously operates in the recirculation mode. The ventilation system is not designed to maintain the ventilated areas pressurized to a positive 1/8" water gauge pressure. If airborne contamination is detected, a high radiation signal (control room recirculation signal) from the recirculation air monitor will start the post-LOCI filter fans which will open their associated gravity discharge dampers, and close the toilet area exhaust duct damper. The post-LOCI filter fan unit inlet dampers are already in the open position. A Unit 1 SIAS A1 and Unit 2 SIAS B1 initiation signal was installed to augment the control room recirculation signal actuation. The SIAS initiation also starts the post-LOCI filter fans and opens their associated gravity discharge dampers, and secures the control room lavatory exhaust fan. Each post-LOCI filter unit is designed to process 10,000 ± 10% cfm of circulated air through HEPA and charcoal filters.

A separate exhaust fan is provided for the lavatory but, during the post-incident period when the air flow is in the complete recirculation mode, the lavatory exhaust is cut off automatically as described above.

All equipment except for ducting is remotely located so as to minimize the fire hazard.

The air conditioning system in this area is divided into three supply and return duct systems: one for each Cable Spreading Room and one for the Control Room. A portion of supply and return air is also routed to the Control Room heating, ventilation, and air conditioning (HVAC) equipment room. Each supply and return branch contains an isolation damper. Smoke detectors are located in the return duct from each zone.

In the event of a fire in one zone, the smoke detector automatically closes the corresponding isolation dampers. The air conditioning system continues to serve the other two zones without interruption.

With the isolation dampers closed, smoke can be evacuated from the isolated zone by means of an auxiliary fan. This fan is selectively connected to the return duct of any zone by operating motorized dampers in the auxiliary duct system. Air from the outside is allowed to enter the supply duct of the isolated zone by operating motorized dampers and manually opening the roof mounted hatch and damper. The operating panel for the motorized dampers and smoke removal fan is located just outside the Control Room entrance in the heater bay area.

<u>Control Room Habitability</u>

In accordance with TMI Item III.D.3.4, "Control Room Habitability," BGE committed to ensure Control Room Operators were adequately protected against the effects

of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions. The NRC concluded that the Control Room habitability systems were acceptable, and that the systems provided safe, habitable conditions within the Control Room under both normal and accident radiation and toxic gas conditions, including LOCAs (Reference 1). This conclusion was predicated upon commitments to install a shield wall to prevent any streaming through the pipe chase into the Control Room from below, and to ensure sufficient self-contained breathing apparatus are available to the Control Room personnel to meet the requirements of Regulatory Guide 1.78.

Subsequently, BGE suspended the Control Room habitability thyroid dose calculation pending the issuance of revised source terms for the evaluation of Control Room habitability from the NRC. The NRC concurred with the adequacy of the associated interim compensatory measures (Reference 2) during this period.

Reference 3 approved the use of an alternative radiological source-term methodology for analyzing design basis accident radiological consequences. The analysis assumptions regarding operation of the Control Room ventilation system are discussed in Section 14. Associated plant modifications are discussed in the appropriate Updated Final Safety Analysis Report sections.

Battery Rooms

The battery rooms, located east of the Cable Spreading Room, are ventilated using air from the access control area. Separate supply and exhaust fans are utilized so as to maintain a negative pressure in these rooms with respect to the surrounding areas. The reserve battery room on Elevation 45'0" is also ventilated by the same supply and exhaust system.

Access Control Area

The access control area common to both plant units is located on Elevation 69'0" and 72'0", partly in the Auxiliary Building and partly in the Turbine Building. In 1984, the original access control area was renovated, and an expansion added in the Turbine Building portion. The access control area is now divided into "Clean" and "Controlled" zones. A separate HVAC system is provided for each zone.

The controlled zone includes hot laboratory, cold laboratory, frisker area, etc. No air is recirculated, all the air is conditioned, and a negative pressure is maintained in the controlled zone.

In the controlled zone, the air is supplied by Access Control HVAC RTU-1 and exhausted by access control exhaust Fans No. 11 and 12 through Unit 2 main exhaust plenum. Access Control HVAC RTU-1 provides filtered, tempered air to the controlled zone. It takes suction on outside air intake through redundant safety-related dampers and discharges air to rooms through duct work to the two gas bottle storage rooms, the hot laboratory, cold laboratory, counting room, clean room, frisker areas, clothing disposal area, and corridors 592 and 594. Note that the redundant safety-related dampers fail closed on a loss of power. A redundant set of radiation monitors will be installed which also allow the dampers to close.

The clean zone includes office, sign in/out, locker room, etc. Negative pressure is not necessarily maintained and the air is recirculated. The clean zone is heated, ventilated, and air conditioned by self-contained units. Access Control HVAC AHU-1 has safety-related dampers that fail closed on a loss of power. A

redundant set of radiation monitors will be installed which also allow the dampers to close.

Main Steam Line Penetration Areas

Heat released by the main steam and feedwater pipes requires that cooling be provided all year round. One cooling and ventilating system is provided for each unit. This system uses outside air as the cooling medium.

Fresh air is mixed with recirculated air as required and supplied through ducting from an air-handling unit located on the floor at Elevation 27'0". The main steam line penetration area for each containment is pressurized and the excess air flows out through the open safety vent to the roof at Elevation 91'6". A room thermostat controls the position of the mixing dampers, which are located upstream of dust-stop filters.

Switchgear Rooms

Redundant and Seismic Category I HVAC and refrigeration systems are provided for each unit, with the exception that the pneumatic tubing is not Seismic Category I. Failure of the tubing does not affect safe shutdown of the plant. The equipment room for both is located on Elevation 69'0" and serves switchgear rooms for both units at Elevation 27'0" and 45'0". These rooms require cooling the year around. An "air conditioning" system supplies filtered air for ventilation and cooling at all times. The HVAC units and refrigeration components are redundant, but the supply and return ducts are not. High temperature air alarms annunciate in the Control Room to signal failure of the HVAC system. One alarm per room provides redundancy since both rooms are supplied by the same HVAC unit. If the unit fails, both rooms will heat up and provide an alarm. The inlet dampers fail open so that the rooms cannot become isolated.

The normal design and operating temperature of the Switchgear Room is 104°F. If both refrigeration units for a Switchgear Room fail, the fans can be arranged to supply 100% outside air to these rooms. The effect of purging with outside air can be evaluated by the operator and appropriate action taken.

Fairbanks Morse Diesel Generator Rooms

The Fairbanks Morse diesel generators are housed in three separate rooms located at Elevation 45'0" in the Auxiliary Building. Heat output from each generator is sufficiently high that cooling must be provided for both summer and winter. The ventilation system for this area is designed to limit room temperature to 120°F in summer and a minimum of 60°F in winter. Outside air is used as the cooling medium. An air-handling unit and mixing box-damper arrangement proportion the outside and recirculated air according to room temperature. When the diesel is running, its room is pressurized and the excess air is forced out through a weatherproof exhaust opening over the outside door. Hot water unit heaters maintain a minimum temperature of 60°F when the diesel is shut down.

Waste Processing Area Ventilation

A common air supply system consisting of three 50% capacity air handling units positioned on the west side of the Auxiliary Building at Elevation 69’0” supplies tempered air for ventilation of the common waste processing area. A system of ductwork ensures a uniform distribution throughout this area.

Separate exhaust systems for Units 1 and 2 draw air from their respective waste processing areas by means of ductwork and force it through HEPA filters, after which it is discharged into the main exhaust plenums provided for each unit. From here, the main plant exhaust fans force the air past the radioactivity monitors and out through the exhaust stacks. These exhaust fans are 100% redundant, but the filters are not.

Emergency Core Cooling Pump Room Ventilation

The ECCS pump rooms for Units 1 and 2 are served by the common waste processing area ventilation supply system. The ECCS pump room exhausts may be directed through HEPA filters prior to emptying into the main plant vent. When the ECCS pumps are operated post-accident, air flow from the ECCS pump room area may be diverted through the charcoal filters by manual remote actuation in the Control Room. However, the operation of this system and the resultant effects on offsite dose calculations are not credited in the accident analysis. This system provides defense-in-depth only.

Fan-coil coolers are installed in each ECCS pump room to provide additional cooling, if necessary, during pump operation.

Spent Fuel Pool Ventilation

An air supply system consisting of two 50% capacity air handling units, located at Elevation 86'0", directly above the three supply units for the waste processing area, provides ventilation for the SFP area. Tempered outside air is supplied to one side of the SFP area at Elevation 69'0". A separate exhaust system picks up air through a manifold, located on the opposite side of the pool, draws it through HEPA filters, and feeds it into the main plant vent of Unit 1. During movement of recently irradiated fuel assemblies, this air may be manually diverted by dampers into charcoal filters after it leaves the HEPA filter bank. Unit heaters are used to maintain a minimum temperature of 60°F in the winter.

The SFP Ventilation System is capable of maintaining a negative pressure with respect to surrounding areas of the building. The limitations placed on this system by the Technical Specifications ensure that, in the event of a fuel handling accident, involving recently irradiated fuel, all radioactive material released from a recently irradiated fuel assembly will be discharged to the atmosphere through the main plant vent. The operation of this system is consistent with the assumptions of the accident analyses.

Auxiliary Feedwater Pump Room

There are "normal" and "emergency" cooling systems used to cool the room. Identical systems are used for both Unit 1 and 2 pump rooms. (Refer to Section 6.9 for a description of the emergency mode of operation.) During normal plant operation, one operable self-contained HVAC unit is capable of maintaining the temperature in this room at 90°F or below. Air for ventilation is drawn in through redundant quick-close dampers from the Turbine Building and is forced out through ducting into the mechanical equipment room at Elevation 5'0" of the Auxiliary Building.

Decontamination Room

This system is intermittent in operation and has by necessity been appended to the normal ventilating system of the Unit 2 side of the Auxiliary Building at Elevation 5'0". While the decontamination room exhaust fan is running, the

"normal" exhaust system from the tank rooms located at the west end of Unit 2 at this level, plus that from the decontamination room itself, is automatically interrupted by means of powered dampers.

By means of a fan located within this room, air is collected from three exhaust hoods which cover separate cleaning areas. It is drawn through a water-pad type scrubber and directed into the normal exhaust system of the Auxiliary Building at this level. Air is supplied by the normal ventilating system to the hallways and enters this room through the connecting passageways. Waste water from the scrubber is directed into the WPS.

Auxiliary Building Ventilation Charcoal Filters

Table 9-19 lists the Auxiliary Building charcoal filters, total flow rates and total charcoal weights. The charcoal is Barnebey-Cheney #727 (or equivalent) impregnated with 5 wt% iodine compounds. The flow velocity through the charcoal bed is 40 fpm in all cases and the corresponding residence time is 0.25 seconds. A typical charcoal filter module is 24-1/4"x25-3/4"x6-1/4". Each module is designed for an air flow of 333 cfm. Each filter housing contains sufficient modules for the total flow rates shown in Table 9-19. Filter testing is explained in Section 6.6.7.

Testing is performed to demonstrate that the installed charcoal adsorbers will perform satisfactorily in removing both elemental and organic iodides for design conditions of flow, temperature, and relative humidity. Periodic testing is conducted to ensure filter efficiencies credited in the accident analysis are maintained.

Diesel Generator Building HVAC System

The HVAC System for the safety-related Diesel Generator Building is divided into safety-related and non-safety-related portions. Two non-safety-related air handling units provide ventilation to the Diesel Generator Building. Air handling unit one (1A-AHU-1) provides ventilation to the Diesel Generator Building Control Room, Battery Room, 1E Switchgear Room, and non-1E Electrical Panel Room. Air handling unit two (1A-AHU-2) provides ventilation to the Maintenance Shop, hallway, Future Expansion Room, and Fuel Oil Storage Tank Room. The Diesel Generator Room is cooled by four safety-related fans when the emergency diesel is in operation and stand-by conditions. While the EDG is not in operation, a non-safety-related ventilation system provides cooling to the Diesel Generator Building Control Room, Battery Room, 1E Switchgear Room, and Non-1E Electrical Panel Room using a constant volume, direct expansion cooling air handler unit one (1A-AHU-1). These rooms are exhausted using a non-safety-related exhaust fan (1A-F-7), except for the battery room which uses a separate safety-related exhaust fan. However, during diesel generator operation, a safety-related supply and exhaust fan provides cooling using only outdoor ambient air. Both the non-safety-related air handling unit one and the safety-related supply and exhaust fan share a common section of the ductwork to supply and exhaust these rooms. Interlocks are provided to ensure that both the safety-related fan and the non-safety-related air handling unit do not operate at the same time. The HVAC system is designed to maintain the diesel generator room between 50º and 120ºF.

Station Blackout Diesel Generator Building HVAC System

The HVAC System for the Station Blackout Diesel Generator Building is designed to maintain the temperature in the Station Blackout Diesel Generator Building within the standards of manufacturers of equipment in the building. Four

augmented-quality fans, each thermostatically controlled, are provided to exhaust air from the Diesel Generator Room. The Station Blackout Diesel Generator Building HVAC System also includes two augmented-quality air handling units to provide ventilation to the building. Air handling unit one (0C-AHU-1) supplies and exhaust conditioned air to the Control Room. Air handling unit two (0C-AHU-2) provides cooling using outside ambient air to provide ventilation to the Fuel Tank Room, Switchgear Room, Diesel Generator Room, Battery Room, and Cable Spreading Area. The Cable Spreading Area, Diesel Generator Room and the Fuel Tank Room are exhausted using an augmented quality exhaust fan (0C-F-6). However, the Battery Room has a separate augmented quality exhaust fan (0C-F-5).

9.8.2.4 Turbine Building

When Units 1 and 2 are both in operation, the Turbine Building requires cooling all year round. Outside air is used for this purpose since it will normally be at least 15 degrees below the Turbine Building maximum design air temperature. Twelve fans, one in each vertical air shaft at Elevation 95'0", supply air from mixing boxes through ducting to all levels of the Turbine Building and heater bay area. The selection of fresh air or recirculated air, which is accomplished by means of dampers located in the mixing box, is manually controlled for each fan by a switch located on the operating floor level. The walls of the vertical shafts are louvered above the fan level, thus permitting them to serve as an intake plenum. All Turbine Building ventilating fans are manually controlled. Plant operations personnel determine the number in operation at any one time.

Exhaust dampers located near the center of each horizontal air shaft are blocked open except as follows.

Horizontal air shafts 3 and 4 (two in center) have two exhaust fans mounted in each shaft in place of the relief dampers to exhaust approximately 80% of the air supplied by the intake fans. The fans are manually controlled. The four exhaust dampers adjacent to these four exhaust fans are blocked shut to prevent exhaust flow back into the building.

A mechanical cooling system is installed to provide local cooling at the four heater drain pump motor locations. Four fan coil units, located on Elevation 27' above each pump motor, are supplied chilled water from two air-cooled water chillers located at the north end of the Turbine Building. Air from the fan units is directed downward over the pump motors by removable ductwork. The fans are manually controlled. The chillers have self-contained controls and are turned on by flow switches in the chilled water lines. The chilled water pump is manually controlled.

Hot water unit heaters maintain a minimum temperature of 60°F at the operating floor level during a shutdown period. Unit heater fans are controlled by thermostats located on the operating floor level.

9.8.2.5 Service Building

The administrative area of the service building is air-conditioned all year round. All administration area rooms have individual temperature control.

The warehouse and shop areas at Elevation 45'0" are ventilated all year round by a single makeup-air unit. The unit provides both mechanical cooling and electric heat for year round operation. Unit heaters supplement the makeup air unit to maintain 60°F minimum in the winter time. Room thermostats control the

operation of the unit heater fans. The lower levels of the service building are ventilated only with fresh air being tempered to 60°F. The two 250 Volt battery rooms (Elevations 12' and 35') are cooled with separate mechanical refrigeration cooling systems.

9.8.2.6 Intake Structure

The amount of heat generated by the saltwater and circulating water pump motors in the intake structure is such that a cooling system is necessary year round. Six air supply units consisting of weather louvers and filter modules capable of high moisture separation efficiency, and a supply fan are located on the six saltwater pump hatches. These supply units force fresh air into the intake structure. Room thermostats control the fans' motors. After absorbing the heat from the saltwater pump motors and the circulating water pump motors, the air exits the intake structure through exhaust vents located on the twelve circulating water pump hatches. These exhaust vents consist of weather louvers capable of high moisture separation efficiency. This system limits the intake structure ambient temperature to approximately 104°F when the outdoor ambient temperature is 95°F. A minimum amount of outside air for ventilation enters through ducts at six locations and is exhausted through ducts by two fans located in the east wall of the service building. Hot water unit heaters prevent freeze-up during periods of complete shut down.

If these air supply units were to fail, the increase in temperature could cause the circulating water pump motors to overheat. Therefore, an over-temperature light and alarm are incorporated into the control board located in the Control Room.

The six fan units in the Intake Structure are not required for continuous satisfactory operation of two saltwater pumps per unit. Natural air circulation, cooling effect from the Intake Structure walls and cooling from the saltwater piping will provide sufficient heat removal for the saltwater pump motors.

9.8.3 CODES AND STANDARDS

The work, equipment and materials conform to the requirements and recommendations of the following codes and standards, as applicable:

National Fire Protection Association Code, Pamphlet 90A
National Electric Code
American Society of Heating, Refrigerating and Air-Conditioning Engineers Guides
Air Moving and Conditioning Association, Inc.
American Society of Mechanical Engineers
Sheet Metal and Air-Conditioning Contractors National Association, Inc.
American Society of Testing Materials

9.8.4 TESTS AND INSPECTIONS

All equipment is accessible for inspection. Testing and performance-indicating instruments are built into each critical apparatus to increase the maintainability and reliability of these systems. Where redundant equipment is provided, it will be operated alternately to provide increased assurance of operability (as required by the Technical Specifications).

In addition, the filters (HEPA and charcoal) are subjected to testing similar to that of the Containment Iodine Removal System filters described in Section 6.7.

9.8.5 REFERENCES

1. Letter from R. A. Clark (NRC) to A. E. Lundvall, Jr. (BGE), dated September 3, 1982, Review of NUREG-0737 Item III.D.3.4, Control Room Habitability
2. Letter from D. G. McDonald, Jr. (NRC) to R. E. Denton (BGE), dated June 22, 1995, Control Room Habitability Interim Analysis for Thyroid Dose
3. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), dated August 29, 2007, Amendment re: Implementation of Alternative Radiological Source Term

TABLE 9-18

PLANT VENTILATION SYSTEM DESIGN CONDITIONS

SYSTEM	TYPE SYSTEM[(c)]	SUMMER (°F) INSIDE	SUMMER (°F) OUTSIDE	WINTER (°F) INSIDE	WINTER (°F) OUTSIDE	MAIN PLANT VENT FLOW CAPABILITY UNIT 1 (cfm)	MAIN PLANT VENT FLOW CAPABILITY UNIT 2 (cfm)
Turbine Building	HV	110	95[(i)]	60	0		
Containment Cooling	AC	120	95[(i)]	60	0		
Pressurizer Compartment	AC	NA	NA	NA	NA		
Cavity Cooling	AC	NA	NA	NA	NA		
CEDM Cooling	AC	NA	NA	NA	NA		
Purge System[(a)]	V	NA	NA	60, 45[(j)]	0	50,000	50,000
Pipe Penetration Rooms[(b)]	V	130	NA	NA	NA	2,000	2,000
Auxiliary Building							
Auxiliary Feedwater Pump Room	HVAC	90	NA	NA	NA		
Control Room	HVAC	75	95[(i)]	75	0		
Cable Room	HVAC	90	95[(i)]	75	0		
Access Control Area						NA	13,900
Health Physicist	HVAC	75	95[(i)]	75	0		
Hot Laboratory	HVAC	75	95[(i)]	75	0		
Other Controlled Rooms	HV	NA	NA	75	0		
Clean Rooms	HV	NA	NA	75	0		
Main Steam Pen. Areas	V	160	95[(i)]	60	0		
Switchgear Rooms	HVAC	104	95[(i)]	104	0		
Diesel Generator Rooms	HV	120	95[(i)]	60	0		
Spent Fuel Pool	HV	110	95[(i)]	60	0	32,000	NA
Radwaste Area	HV	110	95[(i)]	60	0	49,500	49,500
ECCS Pump Room	HVAC	110	95[(i)]	60	0	3,000	3,000
Intake Structure	HVAC	104	95[(i)]	104	0		
Service Building							
Office Area	HVAC	75	95[(i)]	75	0		
Locker Room	HV	NA	95[(i)]	80	0		
Warehouse	HV	110	95[(i)]	60	0		
Shop	HV	110	95[(i)]	60	0		

TABLE 9-18

PLANT VENTILATION SYSTEM DESIGN CONDITIONS

SYSTEM	TYPE SYSTEM[c]	SUMMER (°F) INSIDE	SUMMER (°F) OUTSIDE	WINTER (°F) INSIDE	WINTER (°F) OUTSIDE	MAIN PLANT VENT FLOW CAPABILITY UNIT 1 (cfm)	MAIN PLANT VENT FLOW CAPABILITY UNIT 2 (cfm)
EDG 1A Building							
Battery Room	HVAC	104	95[i]	69	0		
1E Switchgear Room	HVAC	104	95[i]	50[e]	0		
EDG Building Control Room	HVAC	104	95[i]	50	0		
Non-1E Electrical Panel Room	HVAC,V	104	95[i]	50[e][h]	0		
EDG Fan Room	H,V	120	95[i]	50	0		
All Other Rooms Below 3rd Floor	HV	120[d]	95[f,i]	50[e]	0		
Third Floor	V	104[f]	95[i]	0	0		
						136,500	118,400

(a) In operation only when containment is occupied

(b) Operated intermittently as required and during LOCA

(c) H = Heating only
V = Ventilation only
AC = Air Conditioning (cooling) only

(d) The hallway, Maintenance Shop, and Future Expansion Room may reach a maximum temperature of 150°F when the diesel is in operation.

(e) Minimum temperature may be lower in the 1E Switchgear Room, Fuel Oil Storage Tank Room, hallway, Non-1E Panel Room, Maintenance Shop, and the Future Expansion Room during diesel operation under accident conditions concurrent with design basis winter temperatures.

(f) Downstream of the radiators on the third floor, the maximum design temperature may reach 140°F during diesel operation.

(g) Deleted.

(h) 1E Switchgear Room and Non-1E Electrical Panel Room may experience temperatures no lower than 32°F during accident conditions with design basis outside temperatures.

(i) These temperatures reflect the ventilation or AC system design temperature (95°F dry bulb), as recommended in the American Society of Heating Refrigeration and Air Conditioning Engineers Guide of regional design conditions.

(j) Applicable only when Unit 1(2) is in Mode 5 or 6. When defueled, applicable with an RCS vent path of at least 8 in^2 available.

TABLE 9-19

AUXILIARY BUILDING VENTILATION CHARCOAL FILTERS

FILTER NAME AND LOCATION	FLOW RATE (cfm)	APPROX. WT OF CHARCOAL (lbs)
Spent Fuel Pool #11	32,000	4,224
Penetration Room Exh. #11[a]	2,000	264
Penetration Room Exh. #12[a]	2,000	264
Penetration Room Exh. #21[a]	2,000	264
Penetration Room Exh. #22[a]	2,000	264
ECCS Pump Room Exh. #11	3,000	396
ECCS Pump Room Exh. #21	3,000	396
Post-LOCI #11 (Control Room)[a]	10,000	1,500
Post-LOCI #12 (Control Room)[a]	10,000	1,500

[a] These charcoal filter units are tested in accordance with Technical Specification 5.5.11, Ventilation Filter Testing Program.

9.9 FIRE PROTECTION PROGRAM REPORT

9.9.1 INTRODUCTION

The Fire Protection Program at the Calvert Cliffs Nuclear Power Plant (CCNPP) provides the necessary controls to protect the health and safety of CCNPP workers and the general public, satisfy NRC and Insurer requirements, meet applicable State of Maryland codes and standards and safeguard Company assets by preventing fires and minimizing the consequences of any fire that may occur.

The fire protection program is based on NRC requirements and guidelines, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the fire protection program meets the requirements of 10 CFR 50.48(c), which incorporates, with exceptions, the National Fire Protection Association's (NFPA) 805 (Reference 1). The fire protection program also uses the guidance of Nuclear Energy Institute (NEI) 04-02 (Reference 2), as endorsed by Regulatory Guide 1.205 (Reference 3). Adoption of NFPA 805 is the method of satisfying 10 CFR 50.48(a) and General Design Criterion (GDC) 3. Prior to adoption of NFPA 805, draft GDC 3 was followed in the design of safety and non-safety related structures, systems, and components (SSCs), as required by 10 CFR 50.48(a).

NFPA 805 does not supersede the requirements of GDC 3, or 10 CFR 50.48(a). Those regulatory requirements continue to apply. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements are met may be different than under 10 CFR 50.48(b).

An NRC Safety Evaluation was issued (Reference 4) that modified the operating licenses and Technical Specifications to incorporate a new licensing basis in accordance with 10 CFR 50.48(c). Title 10 CFR Part 50, Appendix R is no longer the licensing basis for fire protection at CCNPP.

The licensing basis is a risk-informed, performance-based program based on NFPA 805. This licensing basis requires that CCNPP meet the performance goals, objectives and criteria that are itemized in Chapter 1 of NFPA 805 through the implementation of performance based or deterministic approaches (see Sections 9.9.2.1 and 9.9.2.2). The plant fire protection requirements are established using the methodology in Chapter 2 of NFPA 805, so that the minimum fire protection program elements and design criteria contained in Chapter 3 of NFPA 805 are satisfied. Then the fire areas and fire hazards are established through a plant-wide analysis and a performance based or deterministic approach is applied to meet the nuclear safety performance criteria. As part of the performance based approach, engineering evaluations, probabilistic safety evaluations and the fire modeling calculations are used to show that the criteria are met. Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria. It also specifies that at least one success path to achieve the nuclear safety performance criteria shall be maintained free of damage by a single fire.

A discussion of general and plant specific sections of the fire protection program follow. Refer to Engineering Standard ES-056 (latest revision in FCMS) for Fire Protection Codes and Standards applicable to CCNPP.

Figure 9-22 provides a simplified system drawing of the fire protection system.

9.9.2 INTRODUCTION AND METHODOLOGY

9.9.2.1 Goals and Performance Criteria

The design basis for the fire protection program is based on the nuclear safety and radiological release goals and performance criteria contained in Sections 1.3 and 1.5 of NFPA 805. These goals and performance criteria are described below.

Nuclear Safety Performance Goal

The nuclear safety performance goal is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition. A safe and stable condition is defined as Mode 3 when the reactor vessel head is on and tensioned. In other conditions, safe and stable condition is defined as k_{eff} below 0.99 and coolant temperatures below boiling.

- Nuclear Safety Performance Criteria.

  Fire protection features are capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria is met.

  a. Reactivity Control. Reactivity control is capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity insertion occurs rapidly enough such that fuel design limits are not exceeded.

  b. Inventory and Pressure Control. With fuel in the reactor vessel, head on and tensioned, inventory and pressure control is capable of controlling coolant level such that sub-cooling is maintained such that fuel clad damage as a result of a fire is prevented.

  c. Decay Heat Removal. Decay heat removal is capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition. Feed and bleed methods for decay heat removal are not permitted.

  d. Vital Auxiliaries. Vital auxiliaries are capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

  e. Process Monitoring. Process monitoring is capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.

Radioactive Release Goal

The radioactive release goal is to provide reasonable assurance that a fire will not result in a radiological release that adversely affects the public, plant personnel, or the environment. Structures, systems, and components relied upon to meet the radioactive release criteria are documented in NFPA-805-00004, Radiological Release Review (see FCMS).

- Radioactive Release Performance Criteria.

  Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) is as low as reasonably achievable and does not exceed applicable 10 CFR, Part 20 limits.

9.9.2.2 Methodology

The fundamental fire protection program is established as described in Section 9.9.3. The program includes a fire protection plan, fire prevention, fire brigade, a water supply, standpipes and hose stations, alarm and detection systems, water based fire suppression systems, gaseous fire suppression systems, and passive fire protection features.

Fire areas are established and listed in Table 9-20A. The fire hazards within each fire area are identified. The equipment and cables in each fire area needed to meet the performance criteria in Section 9.9.2.1 are determined. Fire scenarios are defined and modeled for each fire area to evaluate the effects of a fire and fire suppression activities on the ability to achieve the performance criteria. A nuclear safety capability assessment is performed to select systems and equipment, and cables needed to achieve the performance goal in each fire area. The list of systems, equipment and cables selected for the nuclear safety capability assessment are contained in NFPA-805-00005, Nuclear Safety Capability Safety Assessment Methodology Review (B-2 Table). See FCMS. Compliance with the performance criteria is evaluated using either a deterministic or performance based approach. Section 9.9.4 describes these approaches in more detail.

If a performance based approach is used, engineering analyses are performed to demonstrate that performance based requirements are satisfied. These analyses include engineering evaluations, probabilistic safety assessments or fire modeling calculations, as needed. Risk informed changes may be made to the fire protection program in accordance with Unit 1 and Unit 2 License Conditions. Prior NRC review and approval is not required if the change results in a decrease in risk and is consistent with the defense-in-depth philosophy. Additionally, prior NRC review and approval is not required for changes that result in a risk increase of less than or equal to 10^{-7} for core damage frequency (CDF) and 10^{-8} for large early release frequency (LERF).

If a deterministic approach is used, compliance with deterministic criteria (Section 9.9.4) are demonstrated.

The calculations supporting this analysis are plant records in FCMS. Any changes to fire protection program elements (described in Section 9.9.3) are evaluated to determine if they are acceptable. Configuration control is maintained using procedure-required reviews of plant changes. Quality control of the calculational methods is maintained in accordance with procedures.

9.9.3 FIRE PROTECTION PROGRAM AND DESIGN ELEMENTS

Calvert Cliffs Nuclear Power Plant complies with Chapter 3.3, Prevention, of NFPA 805 with approved exceptions. Exceptions are listed within the applicable sections. All exceptions were approved in Reference 4.

9.9.3.1 Introduction

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 of NFPA 805 is documented in NFPA-805-00002, NFPA 805 Chapter 3 Fundamental Fire Protection and Design Elements Review (B-1 Table). See FCMS. A summary of the requirements of NFPA 805 and any exceptions listed in NFPA-805-00002 are discussed below.

Where used in NFPA 805 Chapter 3 the terms “Power Block” and “Plant” refer to structures that have equipment required for nuclear plant operations. Table 9-20A lists the structures and areas that are included in the fire protection program.

9.9.3.2 Fire Protection Plan

A fire protection program is established and is documented in a fleet procedure (CC-AA-211). It describes the fire protection program activities and exceptions including the following aspects of the Fire Protection Plan:

- Management Policy and Direction
- Fire Prevention Activities, including control of combustible materials
- Procedures - The NRC approved (Reference 4) a performance based exception for procedures related to inspection, testing and maintenance of fire protection systems and features credited by the fire protection program. Performance based surveillance frequencies may be updated based on the guidance of Electric Power Research Institute (EPRI) TR-1006756 (Reference 5). As a minimum, surveillance frequencies for fire dampers will be reviewed against the EPRI guidance and updated, if needed.

Administrative controls associated with the Fire Protection Program are provided through several administrative procedures and the Exelon Quality Assurance Topical Report. Administrative requirements provide controls for activities that could affect fire protection, including the following:

- In-situ and transient combustibles;
- Ignition sources;
- Hot work activities;
- Smoking;
- Design, maintenance, and plant modification processes; and
- Surveillance of fire protection systems and equipment.

Fire protection equipment and systems are inspected and tested upon initial installation and periodically thereafter. The inspection and testing is conducted following the guidance of applicable NFPA Codes and Standards as well as recommendations and requirements of the insurance carrier. Plant procedures mandate test frequencies and the testing process. Applicability, compensatory actions, testing requirements, and testing frequencies for those fire protection systems which protect equipment needed to achieve and maintain a safe and stable condition are contained in the CCNPP Technical Requirements Manual (OP-CA-TRM-100). Plant procedures also identify compensatory actions to be taken when equipment required for fire scenario safe and stable actions becomes inoperable.

9.9.3.3 Prevention

9.9.3.3.1 Fire Prevention for Operational Activities

The fire prevention program consists of the necessary elements to address the control of ignition sources and transient combustible materials during all modes of plant operation.

General Activities

- Training on fire safety information is provided for all employees and contractors, including fire prevention, fire reporting and plant emergency alarms.
- Unanalyzed fire hazards that are identified are entered into the corrective action program.
- Administrative controls exist addressing the review of plant modifications and maintenance to ensure that the impact on fire protection and the creation of fire hazards are minimized.

Control of Combustible Material

- Procedures exist for control of general housekeeping and control of transient combustible material.
- Wood used in the power block shall be pressure impregnated or coated with a listed fire-retardant material.
- Plastic sheeting used in the power block shall be fire retardant types that have passed NFPA 701 testing or equivalent.
- Waste and debris shall be removed from an area immediately following completion of work or at the end of a shift, whichever comes first.
- Combustible storage or staging areas shall be established with limits on the types and quantities of material stored there.
- Controls on the use and storage of flammable liquids are in accordance with NFPA 30. Controls on the use and storage of flammable gasses are in accordance with NFPA 50A.

Control of Ignition Sources

- A hot work safety program has been implemented and is in accordance with NFPA 51B.
    - Except that welding, cutting and other hot work in sprinklered buildings is permitted while the suppression system is impaired.
- Smoking is restricted to designated safe areas of the plant.
- Open flames or combustion generated smoke is not permitted for leak or air flow testing.
- Administrative procedures control the use of portable electrical heaters and fuel fired heaters in the power block.

9.9.3.3.2 Structural

Wall, floors, and components required to maintain structural integrity are of noncombustible construction as defined in NFPA 220.

9.9.3.3.3 Interior Finishes

Interior wall or ceiling finishes are in accordance with NFPA 101. Interior floor finishes are in accordance with NFPA 101 for Class I interior floor finishes.

- Interior wall, ceiling and floor finishes for the main Control Room were approved in a 1979 NRC Safety Evaluation (Reference 6). They consist of mineral fiberboard ceiling with aluminum eggcrate inserts, carpet with a less than 25 rating per ASTM E-84, gypsum wallboard and painted concrete masonry unit walls. The auxiliary building, turbine

building, intake structure, containment, and service building are either unfinished or painted.

9.9.3.3.4 Insulation Materials

Thermal insulation materials, radiation shielding, ventilation duct materials and soundproofing materials are noncombustible or limited combustible.

- RTV 627 silicone rubber is used for radiation shielding in various areas. This type of radiation shielding has been tested with the ASTM E-119 surface burn test and withstood exposure for 6 hours.

9.9.3.3.5 Electrical

- Wiring above suspended ceilings shall be kept to a minimum. Where installed, the wiring is listed for plenum use, in armored cable, in metallic conduit or routed in cable trays with solid metal top and bottom covers.
  - The current (circa 2016) configuration of wiring above suspended ceilings has been approved.
- The use of non-metallic or thin wall metallic tubing for power, instrument or control cables has been approved as an acceptable alternative.
- Current (circa 2016) cable installations complied with testing approved by the NRC during evaluation of CCNPP's response to Generic Letter 88-20. New, permanent cable installations comply with acceptable cable construction tests listed in FAQ 06-0022 as described in Attachment 3 of CNG-FES-007 (see FCMS).

9.9.3.3.6 Roofs

Metal roof decking is designed and installed so the roof will not sustain a self-propagating fire when heated from underneath by a fire inside the building. Roof covering is Class A as determined by tests described in NFPA 256.

9.9.3.3.7 Bulk Flammable Gas Storage

- Bulk compressed gas storage is not permitted in buildings with SSCs important to nuclear safety. Storage of flammable gas is located outdoors or in detached buildings so that a fire or explosion will not adversely impact SSCs important to nuclear safety.
- Outdoor high-pressure flammable gas storage tanks are located so that the long axis is not pointed at buildings.
  - A performance based exception was approved by the NRC to allow the current configuration of the H_2 storage tanks.
- Flammable gas storage cylinders not required for normal plant operation are isolated from the system.

9.9.3.3.8 Bulk Storage of Flammable and Combustible Liquids

- Bulk storage of flammable and combustible liquids is not permitted inside buildings containing SSCs important to nuclear safety.
  - A performance based exception was approved by the NRC for the current configuration of the 1A Fuel Oil Storage Tank in the 1A Diesel Generator building. Note that tanks connected to a

system (i.e., day tanks, lube oil tanks) are not considered bulk storage.

9.9.3.3.9 Transformers

- Transformer oil collection basins and drain paths are periodically inspected to ensure that they are free of debris and capable of performing their design function.
  - Transformer oil collection basins are not required for spare transformers (i.e., transformers whose coils are not electrically connected and energized).

9.9.3.3.10 Hot pipes and Surfaces

- Combustible liquids are kept from coming into contact with hot pipes and surfaces, including insulated pipes and surfaces. Administrative controls require the prompt cleanup of oil on insulation

9.9.3.3.11 Electrical Equipment

- Adequate clearance, free of combustible material, is maintained around energized electrical equipment.
  - The definition of “adequate clearance” is contained in Section K.5 of NEI 04-02 (Reference 2)

9.9.3.3.12 Reactor Coolant Pumps

- The RCP oil collection is described in Section 4.1.3.3.2. The NRC had previously approved the existing RCP oil collection system and that approval carried forward as an existing engineering equivalency.

9.9.3.4 Industrial Fire Brigade

9.9.3.4.1 On-Site Fire Fighting Capability

A fully staffed, trained and equipped fire-fighting force is available at all times to control and extinguish fires on site. The force has a minimum compliment of 5 people. Members of the fire brigade do not include the minimum operations shift crew necessary for operation and shutdown of both Units.

- The requirements in NFPA 600, Chapter 5 are met for interior structural fire fighting. CCNPP meets the requirements in NFPA 600, Chapter 5. For exterior fires that could jeopardize the ability to meet the performance criteria described in UFSAR Section 9.9.2.1, the ability to control and extinguish those fires is demonstrated.

No fire brigade member is assigned a task that would require more than a nominal action to put the associated equipment in a safe condition. Additionally, the five assigned fire brigade members are excluded from assignment to the on-shift Emergency Response Organization (other than fire brigade). The on-shift Reactor Operators, Senior Reactor Operators, and Shift Technical Advisor assigned to the Emergency Response Organization are not assigned to the fire brigade.

During every shift, the brigade leader and at least two brigade members have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria. An acceptable alternative is that an Operations

Technical Advisor (a licensed Operator) is dedicated to respond with the fire brigade. If used, Operations Technical Advisor position is in excess of the on-shift Reactor Operators, Senior Reactor Operators, and Shift Technical Advisor assigned to the Emergency Response Organization.

The fire brigade is notified immediately following verification of a fire.

Fire brigade members are required to pass an annual physical examination to determine that they can perform the strenuous activity required during manual firefighting operations. Each fire brigade member is required to be physically fit to wear self-contained breathing apparatus and a respirator.

9.9.3.4.2 Pre-Fire Plans

Current, detailed pre-fire plans are available to the fire brigade for all areas in which a fire could jeopardize the ability to meet the performance criteria described in Section 9.9.2.1. The plans detail the fire hazards in the area, along with any nuclear safety components and fire protection systems that are present. These pre-fire plans are reviewed and updated as needed. They are available in the Control Room. The pre-fire plans also address coordination with other groups during fire emergencies.

9.9.3.4.3 Training and Drills

Training is provided to the fire brigade and other plant personnel commensurate with their emergency responsibilities.

- The fire brigade training complies with the requirements of NFPA 600-2000 Edition. Fire brigade members are given quarterly training and practice in fire fighting, including radioactivity considerations to ensure that each member is thoroughly familiar with the steps to be taken in the event of a fire. A written program details the fire brigade training program. Written records are maintained for each fire brigade member. They include: initial fire brigade classroom and hands on training, refresher training, special training school attendance and leadership training.
- Plant personnel who respond with the fire brigade are trained as to their responsibilities.
- Drills are conducted quarterly for each shift to test the response capability of the fire brigade. Drills are developed to test and challenge fire brigade response, including performance as a team, use of equipment, use of pre-fire plans and coordination with other groups. These drills evaluate the fire brigade's ability to react, respond and demonstrate proper fire fighting techniques. Fire brigade drills are conducted in various plant areas, especially those areas identified to be essential to plant operation and to contain significant fire hazards. Drill records are maintained detailing the drill scenario, fire brigade response and ability of the fire brigade to perform as a team. A critique is held after each drill.

9.9.3.4.4 Fire Fighting Equipment

The fire brigade is provided with approved firefighting protective equipment, including turnout gear and self-contained breathing apparatus. Additional fire-fighting equipment is available, such as: hoses, nozzles, smoke ejectors, foam-making equipment, and other specialized tools. This equipment conforms to the applicable NFPA standards. Minimum

quantities of fire-fighting equipment are identified by plant procedures. Plant procedures also provided for periodic inspection and testing of fire-fighting equipment.

9.9.3.4.5 Off Site Fire Department Interface

Mutual aid agreements have been established with off-site fire departments to provide assistance to the plant fire brigade on an as-needed basis. The mutual aid agreements include an offer of site-specific training for the off-site fire departments and an offer of a plan for interface with the on-site fire brigade. Plant security and radiation protection plans address the response of the off-site fire departments.

9.9.3.4.6 Communications

An effective emergency communications capability is provided to the fire brigade.

9.9.3.5 Water Supply

The fire protection water supply is dedicated for fire protection use only. Except the pre-treated water storage tanks can serve other functions if there is a dedicated capacity capable of providing the maximum fire protection demand. Administrative procedures ensure an adequate water supply in these tanks for fire protection purposes.

Storage Tanks

- The water is supplied by three wells located on site through water pumps to two 500,000 gallon capacity (pretreated) water storage tanks located at the Fire Pump House. The layout of the discharge piping from the tanks is such that a minimum of 300,000 gallons (each tank) is always available to the fire protection system. The 300,000 gallons provides a 2-hour supply for the largest demand suppression system (diesel generator rooms), plus 1,000 gpm available for manual hose streams. Each of the tanks is also equipped with low-level alarms (less than 303,000 gallons) which annunciate in the Control Room and locally. The well pumps have the capacity to replenish the minimum required 300,000 gallons to one of the storage tanks within 8 hours. These tanks can be used as a backup water supply for the auxiliary feedwater system (see Section 10.3.1).
- The pre-treated water tanks are cross connected so that the fire pumps could take suction from either tank. Although the tanks are cross-connected, the cross-connect valve in the fire pump house is maintained locked shut, except when necessary to provide interconnection for firefighting purposes.

Pumps

- Water for the plant fire suppression systems is supplied by two full-capacity fire pumps. One pump is electrically-driven and the other is diesel engine-driven. These pumps are designed and installed in accordance with NFPA 20.
- The diesel engine-driven pump is arranged to provide backup for the electrically-driven pump in case the latter does not start or does not maintain adequate pressure at the header. The diesel engine-driven pump also starts automatically if electric power is interrupted to the electrically-driven pump. The diesel engine-driven fire pump is supplied with 8 hours of fuel from a nominal 500 gallon fuel tank located in the Fire Pump House.

- The NRC approved an exception to the requirement to separate the fire pumps with a rated fire barrier. The fire pumps are in a sprinklered building.
- Both fire pumps are designed to start automatically. The electrically-driven pump starts automatically on a low-header pressure of 95 psig with the diesel engine-driven pump being started at 85 psig.
- There are individual fire pump connections to the yard fire main loop. Each fire pump discharges into the yard main through a 12 inch diameter underground line. An isolation valve is provided between the two points at which the fire pump discharge lines connect to the yard main so that in the event of the failure of one of the pumps, the other pump is still available.
- Excessive pressure developed at the discharge side of the fire pumps is relieved through pressure regulating valves. These valves, along with bypass lines on the wet pipe sprinkler system alarm check valves located in the lower elevations of the plant, prevent over-pressurizing the fire water distribution system.
- A jockey pump is provided to automatically maintain pressure in the system, thus eliminating the need for the main fire pumps to maintain system pressure. This pump maintains a pressure of 115 to 125 psig in the fire protection water system under normal no-use conditions.
- A makeup fire pump is located in a sprinkler area of the Unit 1 Turbine Building basement. The makeup pump takes suction from a plant service water main and discharges to the fire protection system to meet the intermittent usage of water for the purposes other than fire protection. An administrative procedure establishes control for use of the fire system for purposes other than fire-fighting by limiting use to a single 1-1/2 inch hose stream and use of the makeup pump. This restriction to a single 1-1/2 inch hose applies to all non-fire protection use of the fire protection water supply unless evaluated and approved by the site fire protection engineer and documented in a procedure.
- Alarms are provided to immediately notify the Control Room of operation of the fire pumps.

Yard Fire Main, Hydrants and Headers

- An underground yard fire main loop has been designed and installed in accordance with NFPA 24. The fire yard main loop consists of 12 inch cement-lined iron piping and completely surrounds the plant. The yard main is cross-connected by distribution piping that is routed through the plant structures using carbon steel pipe. The distribution piping supplies the various fire protection systems and provides alternate paths for water flow should any portion of the fire main become disabled. The water-based suppression systems in the power block structures and the yard hydrants within the Protected Area of the plant are supplied by this fire protection water supply system. This system also supplies fire protection water to the warehouses.
- A separate fire protection system and yard main encircles the Nuclear Security Facility/Nuclear Office Facility and a single cross-connection is provided to the main plant fire protection system through a normally locked-closed valve. Should the fire protection system inside the Protected Area become disabled, opening this valve permits these fire pumps and associated water supply to provide a back-up to the Protected Area fire protection water supply system. This system is not covered by the plant Quality Assurance program and no credit for its availability is assumed in the plant fire protection design basis.
- Post indicator type valves are provided to isolate portions of the yard fire main loop for maintenance and repair purposes. These isolation devices are located so that isolation of a portion of the yard main loop does not simultaneously shut off the supply to both fixed fire suppression and fire hose stations provided for

manual backup. Sprinkler systems and manual hose station standpipes are connected to the plant fire protection water supply so that a single active failure or a crack in the water supply piping to these systems does not impair both the primary and backup fire suppression systems.

- All hydrants, hose couplings and standpipe risers are threaded so that they are compatible with local fire departments connections.
- Headers fed from each end are permitted inside buildings to supply both sprinkler and standpipe systems. The steel piping and fittings meet the requirements of American National Standards Institute B31.1 up to and including the first valve. The NRC approved an exception to the requirement to seismically design interior supply standpipes and headers. The water supply for firefighting was not required to be designed to withstand a Safe Shutdown Earthquake for these interior parts. These headers are considered an extension of the yard main system. The sprinkler and standpipe system is equipped with an approved shutoff valve.
- The fire protection water supply and fire suppression system control valves are inspected periodically to ensure they are in their correct positions. One of the following methods is acceptable: locking valves in their normal position, sealing valves in their normal position, or providing alarms in the Control Room for valves out of position.
- The requirement is to have fire hydrants which are installed approximately every 250 feet apart on the yard fire main system. Yard hydrants have been provided at intervals of approximately 200 feet to 300 feet around the exterior of the plant. Hose cabinets are equipped with hose, combination nozzle and other auxiliary equipment specified in NFPA 24. These hose cabinets are provided at interval not exceeding 1000 feet along the yard main system.

9.9.3.6 <u>Standpipe and Hose Stations</u>

These systems consist of standpipes/hose stations supplied with water from the fire protection water supply system. The standpipes/hose stations are located throughout the plant in permanent structures. The typical standpipe/hose station system consists of two 2-1/2 inch hose connection outlets. Each of the standpipes/hose stations is also provided with a universal spanner wrench. The standpipes/hose stations are spaced at approximately 100 foot intervals, located on all building elevations, and arranged to reach all safety-related components in the plant.

- For the power block buildings (see Table 9-20A) the NRC approved the use of NFPA 14 Class I standpipe and hose systems in lieu of Class III standpipe and hose systems.
- Hose stations are designed to ensure an adequate water flow rate and nozzle pressure. Hose station pressure reducers are provided where necessary for the safety of the fire brigade and the off-site fire department personnel.
- Hose nozzles supplied to each power block area (see Table 9-20A) are based on the area fire hazards. The combination spray/straight stream nozzle is not used in areas where the straight stream can cause unacceptable damage or present an electrical hazard to fire-fighting personnel. Listed electrically safe fixed fog nozzles are provided at locations where high voltage shock hazards exist. All hose nozzles have shutoff capability and are able to control water flow from full open to full closed.
- Per the requirements of Branch Technical Position 9.5-1, (Reference 7) plants operating in 1977 were not required to provide water to standpipes and hose stations for manual fire suppression in areas containing systems and

components performing nuclear safety functions following a Safe Shutdown Earthquake. This is an exception to NFPA 805.

9.9.3.7 Fire Extinguishers

Portable fire extinguishers are provided at locations throughout the plant. The extinguishing agents utilized are appropriate for the service requirements of the area. The portable fire extinguishers are located and installed following the guidance of NFPA 10. Fire extinguishers may be positioned outside of fire areas due to radiological conditions.

- The fire extinguishers comply with NFPA 10-1970 and 1973 editions. Current (circa 2016) fire extinguisher locations were reviewed per Section K.8 of NEI 04-02 (Reference 2).

9.9.3.8 Fire Alarm and Detection System

Fire Alarm

A fire alarm system is installed in accordance with NFPA 72D for a Class B system, except that the signals are not recorded automatically. The alarm system transmits signals to the Control Room. An audible-visual alarm system is provided in the Control Room with annunciator windows to warn of the occurrence of the following conditions: actuation of any required fire detector, actuation of a fixed suppression system, actuation of a manual fire alarm station, fire-alarm system trouble (includes valve supervision), electrical fire pump operation, diesel fire pump operation, and fire pump trouble. In addition, there are annunciator windows to designate the affected area. An audible alarm which is distinctive from other Control Room alarms is also provided.

Manual pull stations and station communication equipment is installed throughout the power block to allow a person observing a fire at any location to quickly and reliably communicate to the Control Room.

Communications devices and protocols are provided to promptly notify personnel on site, the fire brigade and off-site emergency response agencies of the existence of a fire emergency so they can determine an appropriate course of action. Two independent means of notifying off-site agencies exists.

Detection

A fire detection system consisting of various types of smoke, heat and flame detectors is provided where required as determined by NFPA 805, Chapter 4. These required detection systems are identified in Table B-3 (NFPA-805-00007, see FCMS) and listed in OP-CA-TRM-100. A general description of the types of detection equipment is described below. The required detection systems are installed in accordance with NFPA 72, except smoke detector spacing in the Units 1 and 2 69’ West electrical room (Rooms 529 and 532). This exception is an approved engineering equivalency evaluation.

Smoke detection includes both ionization and photoelectric type detectors. Most of the detectors provide an “alarm only” function, however, there are several smoke detector sub-system installations which also cause actuation of an associated fixed suppression system. A third type of smoke detector, beam type detectors, are installed in the Independent Spent Fuel Storage Installation warehouse and weld shop.

Heat detection consists of both spot-type detectors and line-type detectors. Spot-type detectors consist of one of three types: rate-of-rise, fixed temperature, or a combination of the two types (rate-compensated). The spot-type detectors installed in the plant are generally installed as part of a fixed suppression system. These detectors cause the suppression system to actuate as well as transmit an alarm to the Control Room. The line-type detectors, which are installed in several cable trays in the containment buildings and for some transformers, provide an alarm-only function.

Infra-red type flame detectors are installed in several areas of the Auxiliary Building where smoke detection is not appropriate. The flame detectors provide an alarm-only function.

9.9.3.9 Automatic and Manual Water Based Fire Suppression System

Fixed water suppression systems consist of several different types of systems including deluge systems, pre-action, wet pipe, dry pipe sprinkler, manual, and foam systems. The systems are automatically actuated except for the sprinkler systems protecting the main turbine bearings, the foam systems, and Cable Chase 1A, 1B, 2A, and 2B.

Required water suppression systems are installed in accordance with NFPA 13. Exceptions to sprinkler system installation per NFPA 13 are: the 1B, 2A, and 2B EDG rooms (Rooms 421, 422 and 416), Truck Bay (Room 419), the 1E switchgear room (Room DB104), DG pedestal area (Room SB002), 1A EDG building (various rooms), DG fan room (Room DB203), DG Trench (Room DB004), and underneath the snubber shop (Room 1109).

Each required water suppression system is equipped with a water flow alarm. Alarms from the water suppression systems are annunciated in the Control Room as noted in Section 9.9.3.8. All valves in the required water suppression systems are supervised with one of the following methods: locking valves in their normal position, sealing valves in their normal position, or providing alarms in the Control Room for valves out of position. The required water-based fire suppression systems are equipped with approved isolation valves as described in Section 9.9.3.5.

Deluge water spray system piping is normally dry. These systems are automatically actuated by an associated heat detection system. These systems are installed to provide protection for equipment containing significant quantities of oil. In addition, the systems are provided with open head sprinklers, thus water flows from all the sprinklers upon actuation of the system's deluge valve. The main isolation valve for each of these systems is either electrically supervised or locked in the open position. If the system is actuated, an alarm is automatically transmitted to the Control Room.

Pre-action sprinkler system piping is normally dry. These systems are automatically actuated by an associated heat detection system that allows water into the system piping. In addition, the systems are provided with fusible head sprinklers which operate only when exposed to high temperatures. These systems are installed to provide general area protection (except for the turbine bearing systems which are hazard specific). The main isolation valve for each of these systems is either electrically supervised or locked in the open position. If the system is actuated, an alarm is automatically transmitted to the Control Room.

Wet pipe sprinkler system piping is normally water-filled. These systems are provided with fusible head sprinklers which operate only when exposed to high temperatures. These systems are installed to provide general area protection. The

main isolation valve for each of these systems is either electrically supervised or locked in the open position. If the system is actuated, an alarm is automatically transmitted to the Control Room.

Dry pipe sprinkler system piping is normally dry. These systems are provided with fusible head sprinklers which operate when exposed to high temperatures. These systems are installed to provide general area protection. The main isolation valve for each of these systems is either electrically supervised or locked in the open position. If the system is actuated, an alarm is automatically transmitted to the Control Room.

The foam systems provide protection for the two outdoor fuel oil storage tanks. The foam system piping is normally dry. The systems are designed to be operated manually in the event of a fire. The foam concentrate storage tank is located on the west side of the plant.

9.9.3.10 Gaseous Fire Suppression Systems

These systems are automatically actuated by detection systems located in the protected rooms. Upon actuation, the systems distribute Halon 1301 throughout the Protected Area via system piping. In addition, air flow in and out of the room is isolated prior to system discharge so that the Halon concentration is maintained within the protected room. Actuation of each of these systems is annunciated in the Control Room. Provisions exist for locally disarming the automatic suppression system. These provisions are secured in the operating position and under administrative control.

9.9.3.11 Passive Fire Protection Features

Passive fire protection features include wall, ceiling and floor assemblies, fire doors, fire dampers, and through wall fire penetration seals. Passive features also include electrical raceway fire barrier systems that are provided to protect cables and electrical components and equipment from the effects of a fire.

Building Separation

Each building in the power block is separated from the others by barriers having a designated fire resistance rating of 3 hours or by open space of at least 50 feet, or by a space that meets the requirements of NFPA 80A. The North Service Building and the Turbine Building are treated as a single fire area. An Engineering Equivalency Evaluation justifies excluding the 45’-0” of the North Service Building from the power block. An Engineering Equivalency Evaluation also justifies the nonrated portions of the barriers between the Turbine Building and the Auxiliary Building.

Fire Barriers

Required fire barriers, have fire resistance ratings supported by fire testing, are designated as either 1-hour, 2-hour, or 3-hour fire-rated barriers. The qualification fire tests are performed in accordance with NFPA 251.

The following fire barriers are exceptions to the fire resistance ratings and have been approved using an Engineering Equivalency Evaluation:

- water curtains between the ECCS pump rooms (Rooms 101, 102, 118, and 119) and the containment recirculation pipe tunnels (Rooms 120 and 122);
- gypsum barrier between the east/west hallway of 10’ (Room 104) and the Auxiliary Building stairtower no. 5 (Room AB-5);

- fire area barrier between Unit 1 5' fan room (Room 225) and the Unit 1 27' switchgear room (Room 317);
- blockouts in fire area barriers between Unit 2 Component Cooling room (Room 201) the north/south 5' passage (Room 202 and the Unit 1 Component Cooling room (Room 228);
- blockouts in the fire area barriers between the ECCS pump rooms (Rooms 101, 102, 118 and 119) and the east/west hallway (Room 100);
- blockouts covered with a steel plate installed in fire area barriers on 12'-0" of the Turbine Building;
- gypsum barrier between the Unit 1 69' west Penetration Room (Room 529) and the Spent Fuel Pool/Cask Handling Area (Room 530);
- non-rated wall between the Units 1 and 2 Main Steam Piping Penetration room (Rooms 315 and 309) and the 27'-0" elevation of the Turbine Building (Rooms L27A and L27B);
- lack of fire rated expansion gap seals on elevations 5'-0", 27'-0" and 45'-0" or the Auxiliary Building;
- fire separation between duct banks and cable trays routed above the roof of the Auxiliary Building, Access Control Area and Turbine Building, and the rooms below;
- conduits embedded in the elevation 27'-0" Turbine Building floor slab (Rooms L27A and L27B) and the floor/ceiling slab between stairwells AB-4 and AB-5 and the horizontal cable chases (Rooms 517 and 518);
- non-rated features of the exterior barriers on elevation 45'-0" and 69'-0" of the Auxiliary Building;
- barrier walls between the Charging Pump rooms (Rooms 105A, 105B, 105C, 115A, 115B, and 115C) which are not 3-hour rated; and
- nonrated construction features of exterior fire barriers on various plant buildings.

Fire Barrier Penetrations

Fire-rated penetrations consist of fire doors and fire dampers. Each of the fire-rated doors and dampers are listed for the appropriate fire resistance rating by an independent testing laboratory. Fire barrier seal designs for electrical and piping penetrations were subjected to fire testing at an independent testing laboratory.

Exceptions to the fire resistance ratings are listed below. Water curtains were previously approved by NRC to provide a 3-hour fire separation. The following fire barriers have been approved using an Engineering Equivalency Evaluation:

- 1.5-hour fire rated doors in fire area barriers on elevation 27'-0" of the Auxiliary Building;
- No fire dampers in the fire rated floor slab of the Units 1 and 2 main plant exhaust and equipment rooms (Rooms 524 and 526);
- Lack of a fire damper in the area above the roof of the heater bay and the ALARA Office (Room 571);
- Lack of fire dampers in the barriers separating the Units 1 and 2 DAS computer room (Room 431 and 406) and the Units 1 and 2 Blowdown Tank and piping area (Rooms 428 and 408);
- Non-rated penetration seals in the Auxiliary Building, the North Service Building and the Intake Structure;
- Lack of fire dampers between the following rooms: Unit 1 cable spreading room (Room 306), cable chase 1A (Room 1A), and cable chase 1B (Room 1B);

- 1.5-hour fire rated doors installed in a 3 hour fire barrier in various locations in the Auxiliary Building;
- various deficiencies associated with fire doors are justified in ECP-13-000304;
- tendon access hatches installed in the floor/ceiling on elevations 27'-0", 45'-0" and 69'-0" of the Auxiliary Building;
- pinned open fire dampers throughout the Auxiliary Building;
- doors and non-dampered ventilation penetrations through exterior fire barriers in elevations 45'-0" and 69'-0" of the Auxiliary Building;
- dampers and doors in barrier walls between Charging Pump rooms (Rooms 105A, 105B, 105C, 115A, 115B, and 115C) which are not 3-hour fire rated.

Fire doors comply with NFPA 80, 1970 Edition, except the previously approved watertight doors and bullet proof doors that provide a 3-hour fire separation. Door 113B between No. 11 Charging Pump room (Room 115A) and the No. 13 Charging Pump room (Room 115C), and various deficiencies throughout the plant (ECP-13-000304) are approved exceptions also.

Fire dampers comply with NFPA 90A, 1976 Edition. The following fire dampers have been approved using an Engineering Equivalency Evaluation:

- Dampers between the Control Room (Room 405) and the HVAC equipment room (Room 512) which are not installed per the manufacturer's directions
- Damper installed in the barrier between the Truck Bay (Room 419) and the RC Waste Evaporator room (Room 420), which is not installed per the manufacturer's directions
- Damper between the U1 Plant Exhaust equipment room (Room 524) and the area above the heater bay, which is not installed per the manufacturer's directions
- Dampers in fire barriers in the Auxiliary Building and the Turbine Building which are not installed per the manufacturer's directions and/or are 1.5 hour rated
- Dampers in various locations in the Auxiliary Building which are not installed per the manufacturer's directions

Where penetration seal configurations within the plant are not bounded by a fire test, an engineering evaluation has been performed to document acceptability. The penetration seals in barriers requiring a fire rating are of a commensurate fire rating as the barrier itself.

The annular space between the penetrating item and the opening in a fire barrier is filled with a qualified seal assembly capable of maintaining the fire resistance of the barrier. The following penetrations and sealing material have been approved using an Engineering Equivalency Evaluation:

- Cable tray penetrations in fire barriers in various locations in the Auxiliary Building
- Pipe penetrations through the fire barriers at various locations in the Auxiliary Building
- Pipe penetrations through the fire barriers between the Unit 1 69' West Electrical room (Room 529), the Spent Fuel Pool, Cask Handling Area (Room 530), and the Unit 2 69' West Electrical room (Room 532)
- Drain line penetration through the fire barrier separating the North Service Building from the Intake Structure

- Penetrations through the fire barrier between the Unit 1 69’ West Electrical room (Room 529), and the Spent Fuel Pool, Cask Handling Area (Room 530)
- Penetrations through the fire barrier between (-)10’ East/West hallway (Room 104) and the No. 13 Charging Pump room (Room 115C)
- Use of ceramic fiber for structural penetrations between the Unit 1 69’ West Electrical room (Room 529), the Spent Fuel Pool. Cask Handling Area (Room 530) and the Unit 2 69’ West Electrical room (Room 532)
- Penetrations in the fire barrier between the No. 11 RC Waste Receiver Tank room (Room 114) and the RC Waste Evaporator room (Room 420)
- Use of certain seal designs for Unistrut penetrations throughout the plant (Calculation DE00815)
- Conduits penetrating the fire barrier between the Unit 1 and 2 Battery rooms (Rooms 301, 304, 305 and 307) and the Unit 1 and 2 Cable Spreading rooms (Rooms 302 and 306)
- Cable tray penetrations through fire barriers between the Unit 2 27’ Switchgear room (Room 311), the Unit 2 Containment Purge Air Supply room (Room 312) and the Unit 2 45’ Switchgear room (Room 407)
- Cable bundles penetrating the fire barrier between the Plant Chemistry Data Analysis room (Room 584) and the Plant Chemistry Cold Lab (Room 586)
- Penetrations throughout the plant that can only be inspected from one side (ECP-13-000188)
- Penetrations in the fire barrier between the Charging Pump rooms (Rooms 105A, 105B, 105C, 115A, 115B and 115C) which are not 3-hour fire rated
- Four 1-hour rated link seal penetrations in the 3-hour rated fire barriers in the Auxiliary Building
- Kaowool/Flammastic penetration seal in sleeves penetrated by conduit and cables installed in fire barriers throughout the plant
- Penetration seal design which differs from the typical seal design and is used at various locations throughout the plant (ES199601644-000)

Through penetration fire stops for penetrations such as pipes, conduits, bus ducts, cables, wires, pneumatic tubes and ducts are protected as follows. The annular space between the penetrating item and the opening in the fire barrier is filled with a qualified penetration seal assembly capable of maintaining the fire resistance of the barrier. Conduits are provided with an internal fire seal that has an equivalent fire-rating as that of the barrier. The internal fire seal in conduits is installed on either side of the barrier in a location that is as close to the barrier as possible. An exception to conduit sealing is approved in NFPA 805. Openings in conduit ≤ 4 inches in diameter are sealed with an internal seal unless the conduit extends greater than 5 feet on each side of the fire barrier. In that case, the conduit needs to be provided with non-combustible material to a depth of 2 inches to prevent hot gas and smoke from passing through the fire barrier.

The following penetrations and sealing material have been approved using an Engineering Equivalency Evaluation:

- Typical installation detail used for through penetration fire stops throughout the plant (0113-000092-01)
- Conduits penetrating the fire barrier between the Unit 1 and Unit 2 battery rooms (Rooms 301, 304, 305, and 307) and the Unit 1 and Unit 2 Cable Spreading rooms (Rooms 302 and 306)

- Penetrations in fire barriers throughout the plant that can only be inspected from one side (ECP-13-000188)
- Kaowool/Flammastic penetration seal in sleeves penetrated by conduit and cables installed in fire barriers throughout the plant
- Penetration seal design which differs from the typical seal design and is used at various locations throughout the plant (ES199601644-000)

9.9.4 DETERMINATION OF FIRE PROTECTION FEATURES

Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria outlined above in Section 9.9.2.1. The methodology is permitted to be either deterministic or performance-based. Once a determination has been made that a fire protection system or feature is required to achieve the nuclear safety performance criteria of Section 9.9.2.1, its design and qualification shall meet the applicable requirements of NFPA 805, Chapter 3, as modified by approved changes shown in Section 9.9.3. These fire protection systems and features are documented in NFPA-805-00006, Nuclear Safety Capability Assessment (NSCA) and NFPA-805-00006A, NSCA Analysis Results, both located in FCMS.

<u>Deterministic Approach</u>

Deterministic requirements shall be “deemed to satisfy” the performance criteria, defense-in-depth, and safety margin and require no further engineering analysis.

One success path of required cables and equipment needed to achieve and maintain the nuclear safety performance criteria without the use of recovery actions must be protected. Protection consists of 3-hour encapsulation of the success path, or 1-hour encapsulation of the success path with suppression and detection, or 20 feet separation without intervening combustibles and suppression and detection throughout the area.

In containment, the following protection is needed: a radiant energy shield capable of withstanding a 0.5-hour fire exposure, or 20 feet separation without intervening combustibles, or suppression and detection throughout the area.

Changes may be made to the fundamental fire protection program elements and design requirements if an engineering evaluation demonstrates the alternative is functionally equivalent. Also, changes may be made to fire protection program elements if the alternative is adequate for the hazard as determined by an engineering evaluation. Prior NRC approval is not needed for four specific program element changes: Fire Alarm and Detection Systems, Automatic and Manual Water-Based Fire Suppression Systems, Gaseous Fire suppression systems and Passive Fire Protection Features. Other element changes do require prior NRC approval before they are implemented in accordance with the Unit 1 and Unit 2 License Condition.

<u>Performance Based Approach</u>

The performance based approach is used as an alternative to the deterministic approach described above. The fire scenarios and fire design basis are defined for each fire area considered. These fire areas are evaluated using fire modeling to quantify the fire risk and margin of safety, or by the use of probabilistic safety analysis to examine the impact on overall plant risk.

A fire hazard assessment is performed to quantify the fire risk and margin of safety. To perform a fire hazard assessment, the physical location of the equipment requiring protection is determined using the methods described in Section 9.9.2.2. The damage threshold for this equipment is then established, again using the method described in

Section 9.9.2.2. The fire scenarios are established for this area and the evaluation determines if the equipment needed to achieve the nuclear safety performance criteria are maintained free of fire damage. There must be sufficient margin between the maximum expected fire scenario and the limiting (deterministic) fire scenario.

As an alternative, the overall plant risk of the fire protection approach for a fire area may be evaluated. The fire risk evaluation compares the risk associated with the implementation of deterministic requirements with a proposed alternative. The difference in risk of the two approaches must meet a CDF of 10^{-4}/yr and a LERF of 10^{-5}/yr per Regulatory Guide 1.174 (Reference 8). Additionally, prior NRC review and approval is not required for individual changes that result in a risk increase of less than or equal to 10^{-7}/yr for CDF and 10^{-8}/yr for LERF. These criteria were approved in Reference 4 and are contained in the station license conditions. The proposed alternative must also ensure that the philosophy of defense in depth and a sufficient safety margin are maintained.

<u>Operations Guidance</u>

Guidance is provided to the plant operators that details the credited success paths for each fire area, including the performance of recovery actions and repairs. Recovery actions credited to achieve the nuclear safety performance criteria are feasible. Station Abnormal Operating Procedures contain this guidance. If command has shifted from the Control Room to the alternate shutdown panel, the actions taken at the alternate shutdown panel are not considered to be recovery actions.

9.9.5 REFERENCES

1. NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition
2. NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c), Revision 2, February 2006
3. Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, Revision 1, December 2009
4. Letter from R. V. Guzman (NRC) to B. C. Hanson (EGC), dated August 30, 2016, Issuance of Amendments Regarding Transition to a Risk-Informed, Performance Based Fire Protection Program in Accordance with 10 CFR 50.48(c)
5. EPRI Technical Report 1006756, Fire Protection Equipment Surveillance Optimization and Maintenance Guide, July 2003
6. Letter from R. W. Reid (NRC) to A. E. Lundvall (BGE), dated September 14, 1979, Issuance of Amendments 41 and 23 to Calvert Cliffs Nuclear Power Plant
7. Branch Technical Position APCSB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976, February 24, 1977
8. Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2, May 2011

TABLE 9-20A

POWER BLOCK STRUCTURES AND FIRE AREA DESCRIPTION

Power Block Structure	Fire Area	Description
Auxiliary Building	1	U2 21 ECCS Pump Room
Auxiliary Building	2	U2 22 ECCS Pump Room
Auxiliary Building	3	U1 12 ECCS Pump Room
Auxiliary Building	4	U1 11 ECCS Pump Room
Auxiliary Building	5	11 Charging Pump Room
Auxiliary Building	6	12 Charging Pump Room
Auxiliary Building	7	13 Charging Pump Room
Auxiliary Building	8	22 Charging Pump Room
Auxiliary Building	9	23 Charging Pump Room
Auxiliary Building	10	(-) 10'/(-) 15' Hallways and General Areas/ 21 Charging Pump Room
Auxiliary Building	11	Auxiliary Building (All elevations) General and Misc Areas
Auxiliary Building	12	U2 Component Cooling Room
Auxiliary Building	13	U2 5' Fan Room
Auxiliary Building	14	U1 5' Fan Room
Auxiliary Building	15	U1 Component Cooling Room
Auxiliary Building	16	U1 Cable Spreading Room and 1C Cable Chase
Auxiliary Building	16A	11 Battery Room
Auxiliary Building	16B	Hallway Outside U1 CSR and Battery Rooms
Auxiliary Building	16C	12 Battery Room
Auxiliary Building	17	U2 Cable Spreading Room and 2C Cable Chase
Auxiliary Building	17A	21 Battery Room
Auxiliary Building	17B	Hallway Outside U2 CSR and Battery Rooms
Auxiliary Building	17C	22 Battery Room
Auxiliary Building	18	U2 27' Switchgear Room
Auxiliary Building	18A	U2 Containment Purge Air Supply Room
Auxiliary Building	19	U1 27' Switchgear Room
Auxiliary Building	19A	U1 Containment Purge Air Supply Room
Auxiliary Building	20	Cable Chase 1A
Auxiliary Building	21	Cable Chase 1B
Auxiliary Building	22	Cable Chase 2A
Auxiliary Building	23	Cable Chase 2B
Auxiliary Building	24	Control Room Complex
Auxiliary Building	25	U2 45' Switchgear Room
Auxiliary Building	26	U2 45' E Electrical Pen Room
Auxiliary Building	27	U2 45' W Electrical Pen Room
Auxiliary Building	28	2B Diesel Generator Room
Auxiliary Building	29	U2 RWT Room
Auxiliary Building	30	1B Diesel Generator Room and RC Waste Room
Auxiliary Building	31	2A Diesel Generator Room
Auxiliary Building	32	U1 45' W Electrical Pen Room
Auxiliary Building	33	U1 45' E Electrical Pen Room
Auxiliary Building	34	U1 45' Switchgear Room
Auxiliary Building	35	U2 Horizontal Cable Chase

TABLE 9-20A

POWER BLOCK STRUCTURES AND FIRE AREA DESCRIPTION

Power Block Structure	Fire Area	Description
Auxiliary Building	36	U1 Horizontal Cable Chase
Auxiliary Building	37	U1 69’ W Electrical Pen Room
Auxiliary Building	38	U2 69’ W Electrical Pen Room
Auxiliary Building	39	U1 Service Water Pump Room
Auxiliary Building	40	U2 Service Water Pump Room
Auxiliary Building	41	Misc Waste Evap Control Panel Room
Auxiliary Building	42	U1 AFW Pump Room
Auxiliary Building	43	U2 AFW Pump Room
Auxiliary Building	44	U1 RWT Pump Room
Auxiliary Building	AB-1	Aux Bldg Stairtower No. 1
Auxiliary Building	AB-2	Aux Bldg Stairtower No. 2
Auxiliary Building	AB-3	Aux Bldg Stairtower No. 3
Auxiliary Building	AB-4	Aux Bldg Stairtower No. 4
Auxiliary Building	AB-5	Aux Bldg Stairtower No. 5
Auxiliary Building	ABFL	Aux Bldg Slab Containing NFPA 805 Embedded Conduits 69’
Containment – Unit 1	1CNMT	U1 Containment
Containment – Unit 2	2CNMT	U2 Containment
1A Emergency Diesel Building	DGB1	1A Diesel Generator Building
0C Station Blackout Diesel Generator Building	DGB2	0C Diesel Generator Building
Intake Structure	IS	Intake Structure
Auxiliary Building	KWAL	Vertical K-Line Wall containing NFPA 805 Embedded Conduits
Turbine Building/North Service Building (12’ and 27’ Elevations)	TBFL	Turbine Bldg Slab Containing NFPA 805 Embedded conduits 12’
Turbine Building/North Service Building (12’ and 27’ Elevations)	TB/NSB/ACA	U1 and U2 Turbine Building, North Service Building, Access Control Area
13.8 kV Switchgear House Unit 1	YARD	Outside Yard Area and Buildings
13.8 kV Switchgear House Unit 2	YARD	Outside Yard Area and Buildings
Condensate Storage Tank No. 12 Enclosure	YARD	Outside Yard Area and Buildings
Fire Protection Pump House	YARD	Outside Yard Area and Buildings
No. 2 Fuel Oil Storage Tank No. 21 Building	YARD	Outside Yard Area and Buildings
Pretreated-Well Water House	YARD	Outside Yard Area and Buildings

TABLE 9-20

DESIGN DATA FOR FIRE PROTECTION SYSTEM COMPONENTS

Fire Pump, Electrically-Driven

Type	Horizontal Centrifugal
Number	1
Capacity	2,500 gpm
Discharge Press.	125 psig
Material:	
Discharge Head	Cast Iron
Impeller	Bronze
Motor	250 hp/460 Volts/3 phase/60 Hz
Codes	U.L. Label Motor: NEMA Pump: Standards of the Hydraulic Institute

Fire Pump, Diesel Engine-Driven

Type	Horizontal Centrifugal
Number	1
Capacity	2500 gpm
Discharge Press.	125 psig
Material:	
Discharge Head	Cast Iron
Impeller	Bronze
Engine	283 hp
Codes	Pump: Standards of the Hydraulic Institute

Fire System Jockey Pump

Type	Horizontal Centrifugal
Number	1
Capacity	30 gpm
Discharge Press.	129 psig
Material:	
Discharge Head	Cast Iron
Impeller	Bronze
Motor	7-1/2 hp/460 Volts/3 phase/60 Hz
Codes	Motor: NEMA Pumps: Standards of the Hydraulic Institute

Makeup Pump, Electrically-Driven

Type	Horizontal Centrifugal
Number	1
Capacity	215 gpm
Discharge Press.	125 psig
Material:	
Discharge Head	Cast Iron
Impeller	Bronze
Motor	40 hp/460 Volts/3 phs/60 Hz
Codes	Motor: Underwriters Label, NEMA Pumps Standards of the Hydraulic Institute

TABLE 9-20

DESIGN DATA FOR FIRE PROTECTION SYSTEM COMPONENTS

Piping, Fittings and Valves

	Underground	**Aboveground**
Material	Cast Iron	Carbon Steel[a]
Design Pressure	150 psig	175 psig
Design Temperature	100°F	100°F
Construction	Mechanical Joint	Welded and Screwed
Valves	Cast Iron	Cast Iron[a][c]
	Mechanical Joint	Flanged[b]
	175 psi	175 psi
	UL Label	UL Label

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[a] For 3-1/2" and smaller size fittings, and for 2" and smaller size valves, alternate materials can be used.

[b] 2" and smaller valves are screw type.

[c] For 2-1/2" and greater size gate valves, material may be Ductile Iron and have a pressure class of up to 250 psi.

9.10 COMPRESSED AIR SYSTEM

9.10.1 DESIGN BASIS

The Compressed Air System consists of the instrument air and plant air subsystems. The instrument air subsystem is designed to provide a reliable supply of dry and oil-free air for the pneumatic instruments and controls and pneumatically operated containment isolation valves. The plant air subsystem is designed to meet necessary service air requirements for plant maintenance and operation. The designs of each subsystem are based on an estimated instrument air requirement of 260 scfm and an estimated plant air requirement of 600 scfm. The instrument air subsystem compressor is sized for 450 scfm.

9.10.2 SYSTEM DESCRIPTION

The Compressed Air System is shown schematically on Figures 9-23 (Unit 1) and 9-28 (Unit 2). The Plant Water and Air Service System is shown in Figure 9-29.

The system incorporates two full-capacity, non-lubricated compressors for instrument air, each having a separate inlet filter aftercooler and moisture separator. The instrument air compressors then discharge to a single header which is connected to two air receivers. Both air receivers discharge to a compressed air outlet header which supplies instrument air to the air dryers and filter assembly. The compressed air header then divides into branch lines supplying the pretreatment and tank storage area, the Intake Structure, the service building, the water treatment area, the Turbine Building, the containment structure, and the Auxiliary Building.

An emergency back-up tie from the plant air header has been provided to automatically supply air to the instrument air system if the pressure to the instrument filter and dryer assembly falls below a preset value. Local controls are provided to prevent plant air use when this occurs. For the transition from normal to emergency service, air storage tanks provide an approximate 20-minute supply (Table 9-21).

Particle size, dew point, and oil hydrocarbons are controlled for instrument air supply in accordance with Instrument Society of America standards. Additionally, the Calvert Cliffs approach to controlling air quality was submitted to the NRC in response to Generic Letter 88-14.

One full-capacity plant air compressor with an inlet filter, and integral air coolers and moisture separators, discharges to the plant air receiver. The receiver outlet header is connected to the prefilter assembly, which is followed by an outlet header branching into two separate air headers, one to the instrument air dryers and filter assembly, and the other to the plant air pretreatment and storage tank area, the Intake Structure, the service building, the water treatment area, the Turbine Building, the Containment Structure, and the Auxiliary Building. A system cross-tie between Unit 1 and Unit 2 has been provided for the plant air headers. Additionally, each plant air system has a permanent connection for the installation of a portable air compressor to allow for maintenance of the compressors or SRW system during Modes 3, 4, 5, 6 and defueled. This connection may also be used in Modes 1 and 2 to provide a contingency backup to an operating plant air compressor should the other installed plant air compressor be unavailable.

9.10.3 SYSTEM COMPONENTS

Ratings and construction of system components are listed in Table 9-21.

9.10.4 SYSTEM OPERATION

A continuous supply of instrument air is provided to hold various pneumatically-operated valve actuators in the positions necessary for operating conditions. Normally, the plant air

compressor and one instrument air compressor will operate and the second instrument air compressor will be on automatic standby.

9.10.5 SYSTEM RELIABILITY

The power supply for the normal compressors is the normal distribution system and can be backed up by the EDG. Additional emergency air compressors, known as the saltwater air compressors (SWACs), provide redundant air supply to most safety-related components when the normal air compressors are lost. The SWACs (Table 9-16B) are seismically qualified, air-cooled, and oil-free. The instrument air portion of the compressed air system is primarily used for valve actuation and is not used in any reactor indication, control, or protective circuitry. These valve actuators are designed to fail in the safe position after loss of the instrument air supply. The design of the system and installed equipment redundancy ensure that total loss of instrument air supply is highly improbable. Concurrently, attention has been given to ensure that valve failures from loss of instrument air supply are consistent with the capability to maintain the plant in a safe condition and mitigate the consequences of any simultaneous incident or accident.

9.10.6 TESTS AND INSPECTIONS

Each component is inspected and cleaned prior to installation into the system. Instruments were calibrated during testing and automatic controls were tested for actuation at the proper setpoints. Alarm functions were checked for operability and limits during plant operational testing. The systems were operated and tested initially with regard to flow paths, flow capacity, and mechanical operability.

TABLE 9-21

COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION

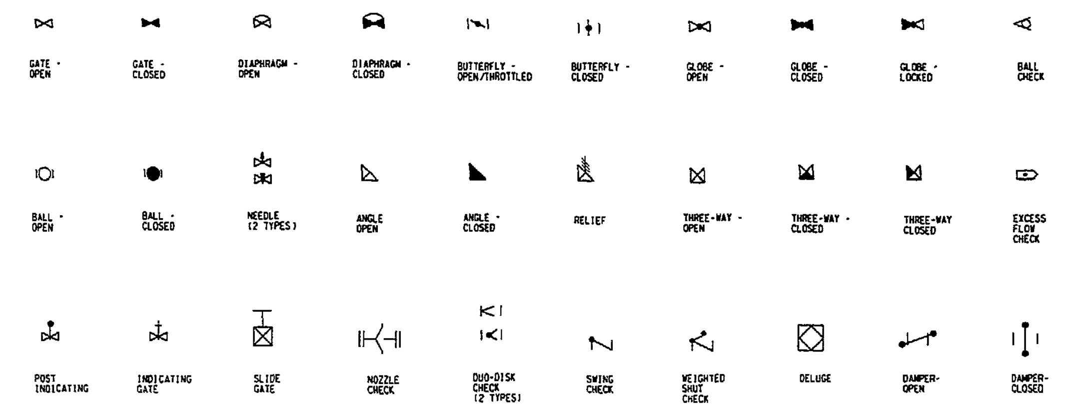
A. INSTRUMENT AIR SYSTEM

Air Compressor	
Type	Vertical, non-lubricated reciprocating, two state Y-angle type
Quantity	2 (per unit)
Design capacity (scfm)	470 (each)
Discharge pressure (psig)	100
Motor	100 hp, 3 phase, 60 Hz, 460 Volt
Code	ASME Section VIII, NEMA
Intake Filter – Silencer	
Type	dry
Quantity	2 per Unit
Base size	8"
Aftercooler and Moisture Separator	
Type	Shell and tube
Quantity	2 (1 per compressor)
Code	TEMA Class C, ASME Section VIII
Air Receiver	
Type	Vertical
Quantity	2 (1 per compressor)
Design pressure (psig)	115
Actual volume (ft^3)	96
Code	ASME Section VIII
Prefilters	
Type	Cartridge
Quantity	2 per Unit
Capacity (scfm)	720
Filtration	99% removal of all liquids, oil, and water droplets
Air Dryer	
Type	Heatless
Desiccant	Activated alumina absorbent
Quantity	2 per unit
Capacity (scfm)	475 (Nos. 12 and 22), 700 (Nos. 11 and 21)
Outlet moisture content with saturated air inlet	-40°F dew point at 100 psig
Afterfilters	
Type	Cartridge
Quantity	2 per Unit
Capacity (scfm)	600
Filtration	100% removal of all particulates over 0.9 microns

TABLE 9-21

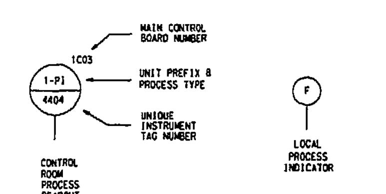
COMPRESSED AIR SYSTEM COMPONENT DESCRIPTION

<u>Piping and Valves</u>	
Valves	150 psi ANSI for 2-1/2" and larger, 600 psi ANSI for 2" and smaller
Piping	Seamless ASTM A106, Grade B (2-1/2" through 24")
Code	ANSI B31.1 (ANSI B31.7 - penetration piping)
B. PLANT AIR SYSTEM	
<u>Air Compressor</u>	
Type	Centrifugal, two stage, with integral air coolers and moisture separators
Quantity	One per Unit
Design capacity (scfm)	600
Discharge pressure (psig)	100
Motor	200 hp, 3 phase, 60 Hz, 460 Volt
Code	NEMA
<u>Intake Filter Silencer</u>	
Type	Dry
Quantity	One per Unit
<u>Air Receiver</u>	
Type	Vertical
Quantity	1
Design pressure (psig)	115
Actual volume (ft^3)	96
Code	ASME Section VIII
<u>Prefilter</u>	
Type	Cartridge
Quantity	2 per Unit
Capacity (scfm)	720
Filtration	99% removal of all liquids, oil, and water droplets
<u>Piping and Valving</u>	
Valves	150 psi ANSI for 2-1/2" and larger, 600 psi ANSI for 2" and smaller
Piping	Seamless ASTM A106, Grade B (2-1/2" through 24")
Code	ANSI B31.1 (ANSI B31.7 - penetration piping)
C. INSTRUMENT BACKUP AIR SYSTEM	
<u>Storage Tank</u>	
Type	Vertical
Quantity	4
Capacity	300 ft^3
Design pressure (psig)	225
Code	ASME Section VIII
<u>Air Amplifier</u>	
Ratio	2:1



VALVE TYPES

INLINE/MISCELLANEOUS COMPONENTS



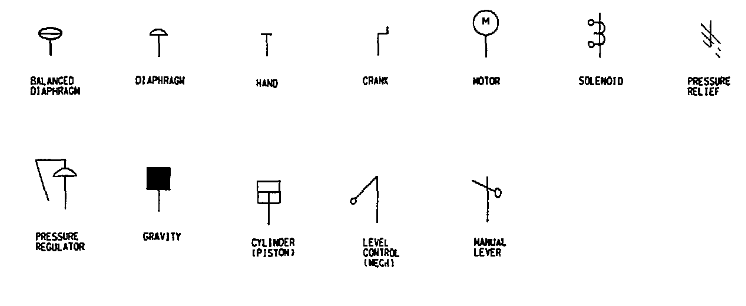
INSTRUMENTATION

0 - COMMON
1 - UNIT 1
2 - UNIT 2
T - TEMPERATURE
P - PRESSURE
F - FLOW
S - SPEED
V - VIBRATION
I - INDICATOR
R - RECORDER
Q - INTEGRATOR
C - CONTROLLER
E - ELEMENT
O - ORIFICE
A - ANALYZER
L - LEVEL
dP - DIFFERENTIAL PRESSURE
MOV - MOTOR OPERATED VALVE
MO - MOTOR OPERATOR
PO - PISTON OPERATOR
RV - RELIEF VALVE
PCV - PRESSURE CONTROL VALVE
TCV - TEMPERATURE CONTROL VALVE

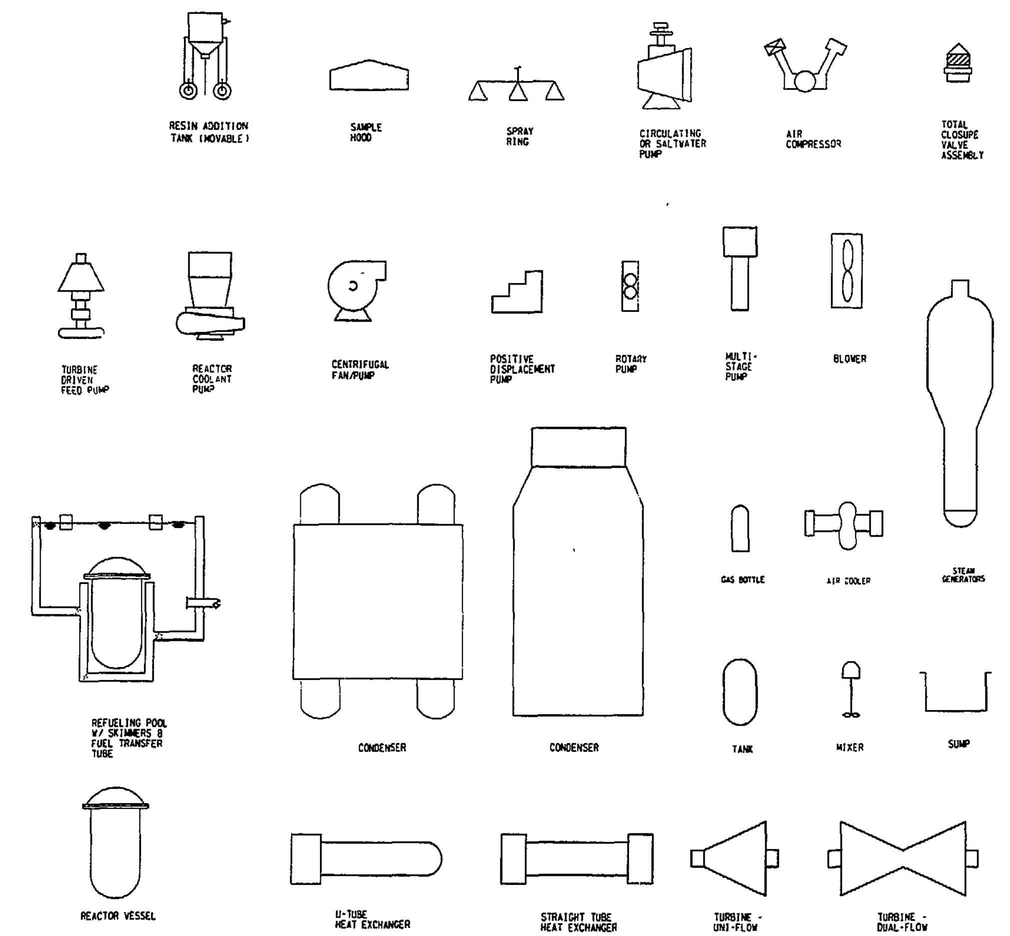
CA - CHEMICAL ADDITION
PS - PRIMARY SAMPLING
BD - BLOWDOWN
DCW - DIESEL COOLING WATER
DFO - DIESEL FUEL OIL
DSA - DIESEL STARTING AIR
DLO - DIESEL LUBE OIL
SS - SECONDARY SAMPLING
MS - MAIN STEAM
ES - EXTRACTION STEAM
CD - CONDENSATE
FW - FEEDWATER
AFW - AUXILIARY FEEDWATER
PA - PLANT AIR
IA - INSTRUMENT AIR
SW - SALTWATER
WBP - WATERBOX PRIMING
CAR - CONDENSER AIR REMOVAL
RDV - REHEATER DRAINS AND VENTS
RE - RADIATION ELEMENT
DR - STEAM DRAIN
pH - HYDRONIUM CONCENTRATION
SO - SOLENOID OPERATOR

SRW - SERVICE WATER
RC - REACTOR COOLANT
CVC - CHEMICAL & VOLUME CONTROL
SI - SAFETY INJECTION
WGS - WASTE GAS SYSTEM
MWS - MISCELLANEOUS WASTE SYSTEM
RCW - REACTOR COOLANT WASTE
CC - COMPONENT COOLING
FP - FIRE PROTECTION
SFP - SPENT FUEL POOL COOLING
AHB - AUXILIARY HEATING BOILER
N2 - NITROGEN
H2 - HYDROGEN
O2 - OXYGEN
CW - CIRCULATING WATER
PSW - PLANT SERVICE WATER
DW - DEMINERALIZED WATER
HVAC - HEATING, VENTILATION & AIR CONDITIONING
SWP - SOLID WASTE PROCESSING
SX - SAMPLE CONNECTION
CV - CONTROL VALVE
BTV - BLEEDER TRIP VALVE

ABBREVIATIONS



ACTUATORS



EQUIPMENT

NOTES

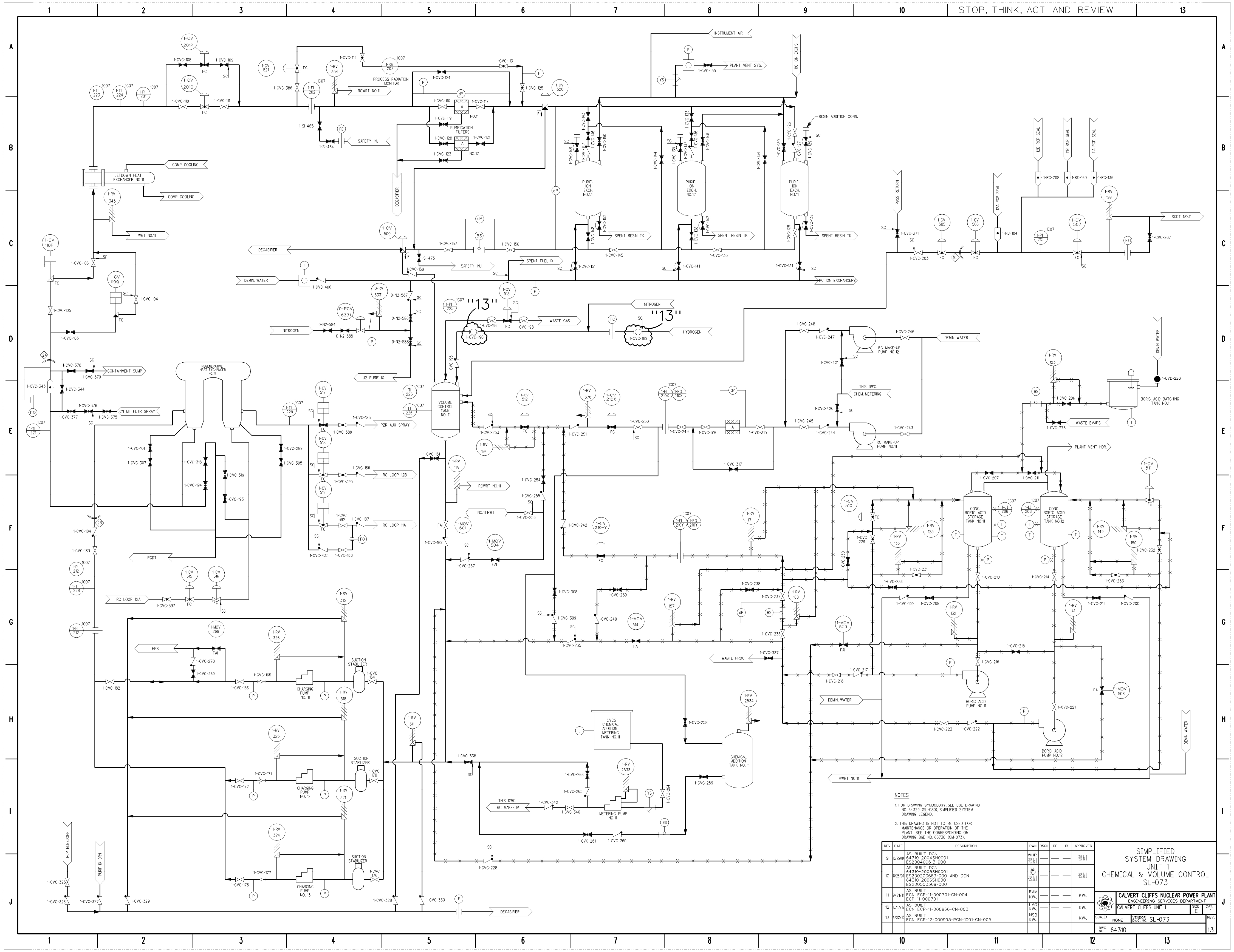
1. THIS DRAWING IS THE SYMBOL LEGEND FOR THE SIMPLIFIED SYSTEM DRAWINGS ONLY.

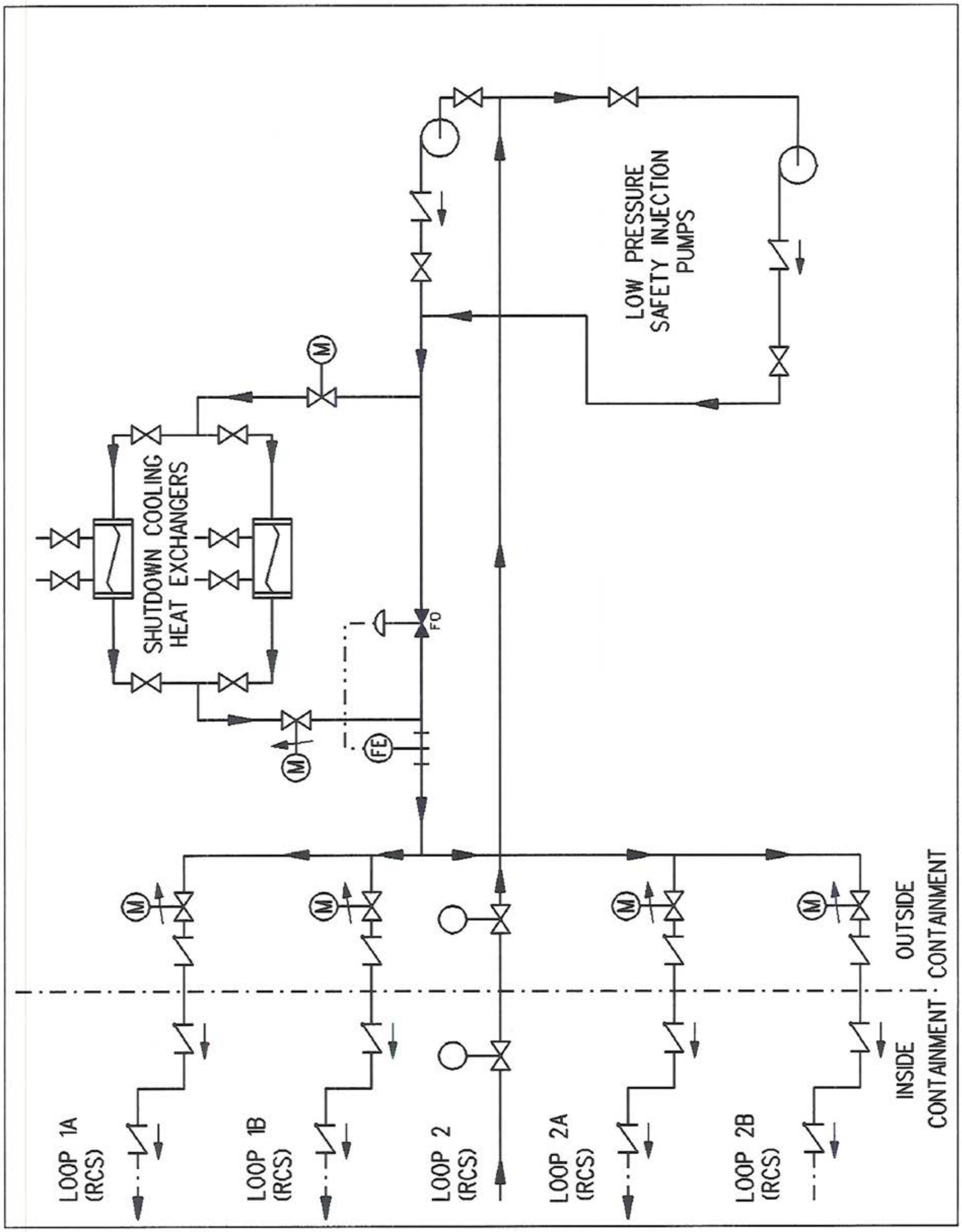
2. THIS DRAWING IS NOT TO BE USED FOR MAINTENANCE OR OPERATION OF THE PLANT. SEE THE CORRESPONDING [illegible] DRAWING, BGE NO. [illegible] ([illegible]).

CALVERT CLIFFS NUCLEAR POWER PLANT
UFSAR FIGURE 9-1
LEGEND
BGE DRAWING 64-329, REV 1

Revision 21

FIGURE 9-3 CHEMICAL AND VOLUME CONTROL SYSTEM – UNIT 1





Calvert Cliffs Nuclear Power Plant

SHUTDOWN COOLING FLOW DIAGRAM

Figure 9-5

Revision 45

FIGURE 9-6 COMPONENT COOLING SYSTEM – UNIT 1

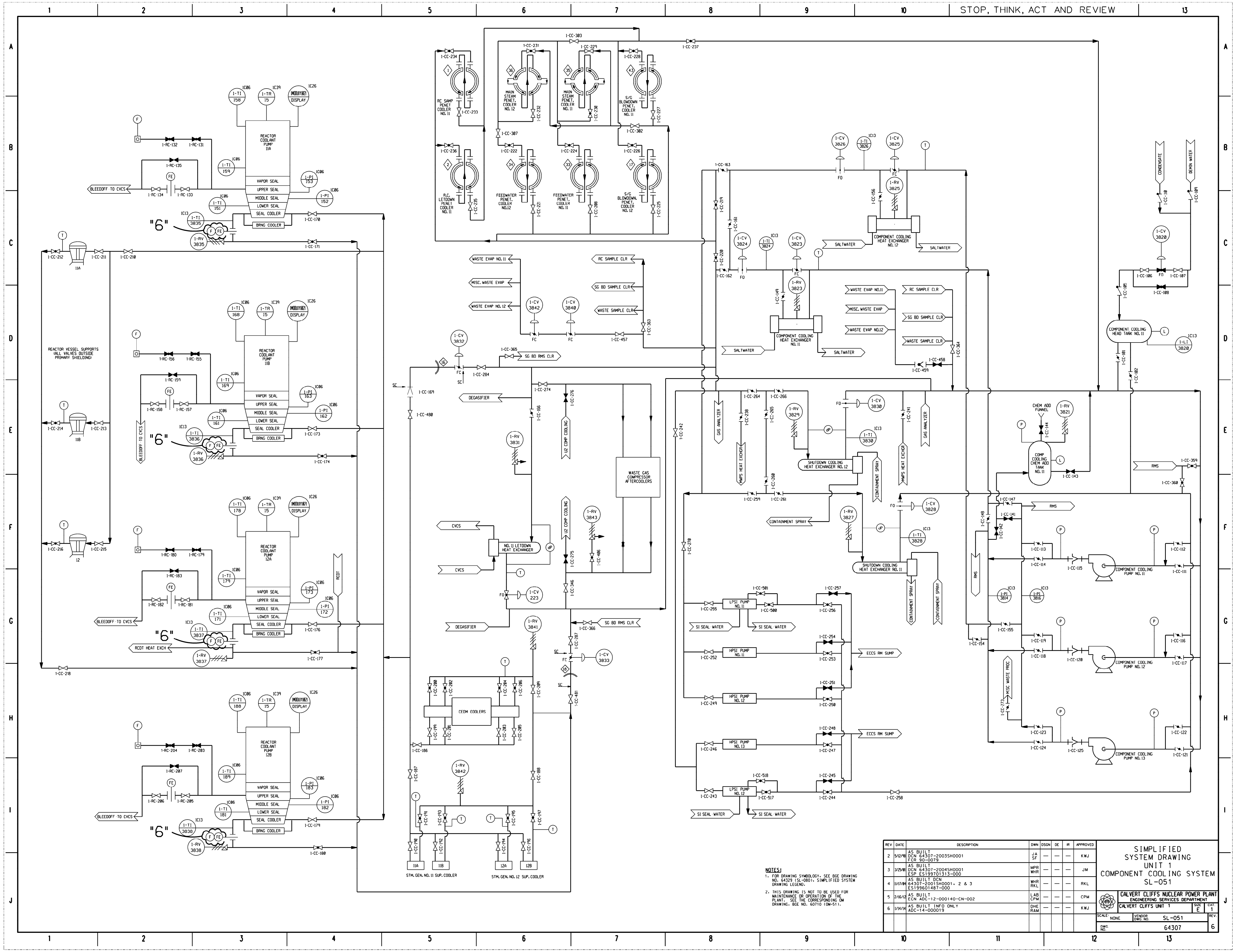


FIGURE 9-7 SPENT FUEL POOL COOLING

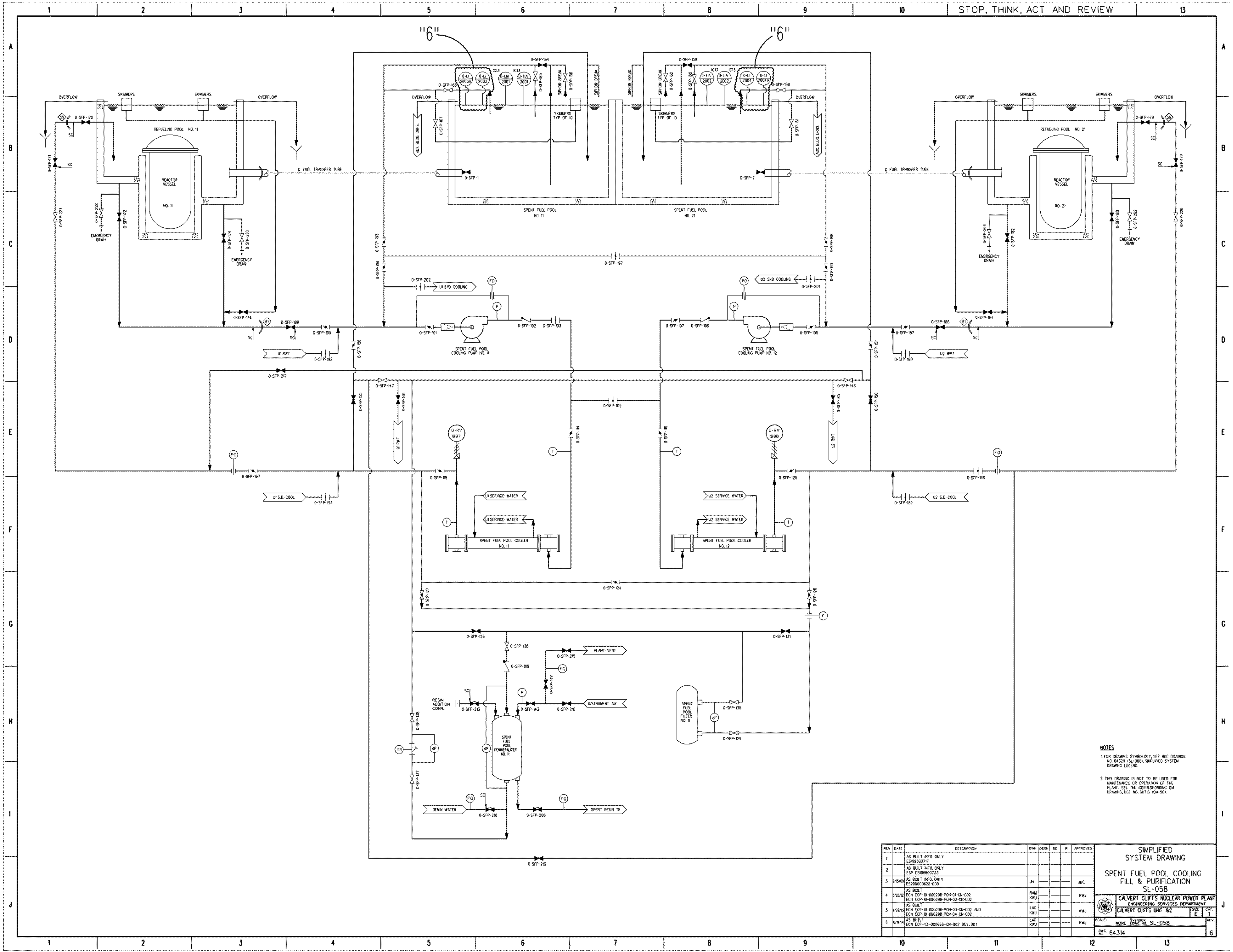


FIGURE 9-8 CIRCULATING AND SALTWATER COOLING SYSTEM – UNIT 1

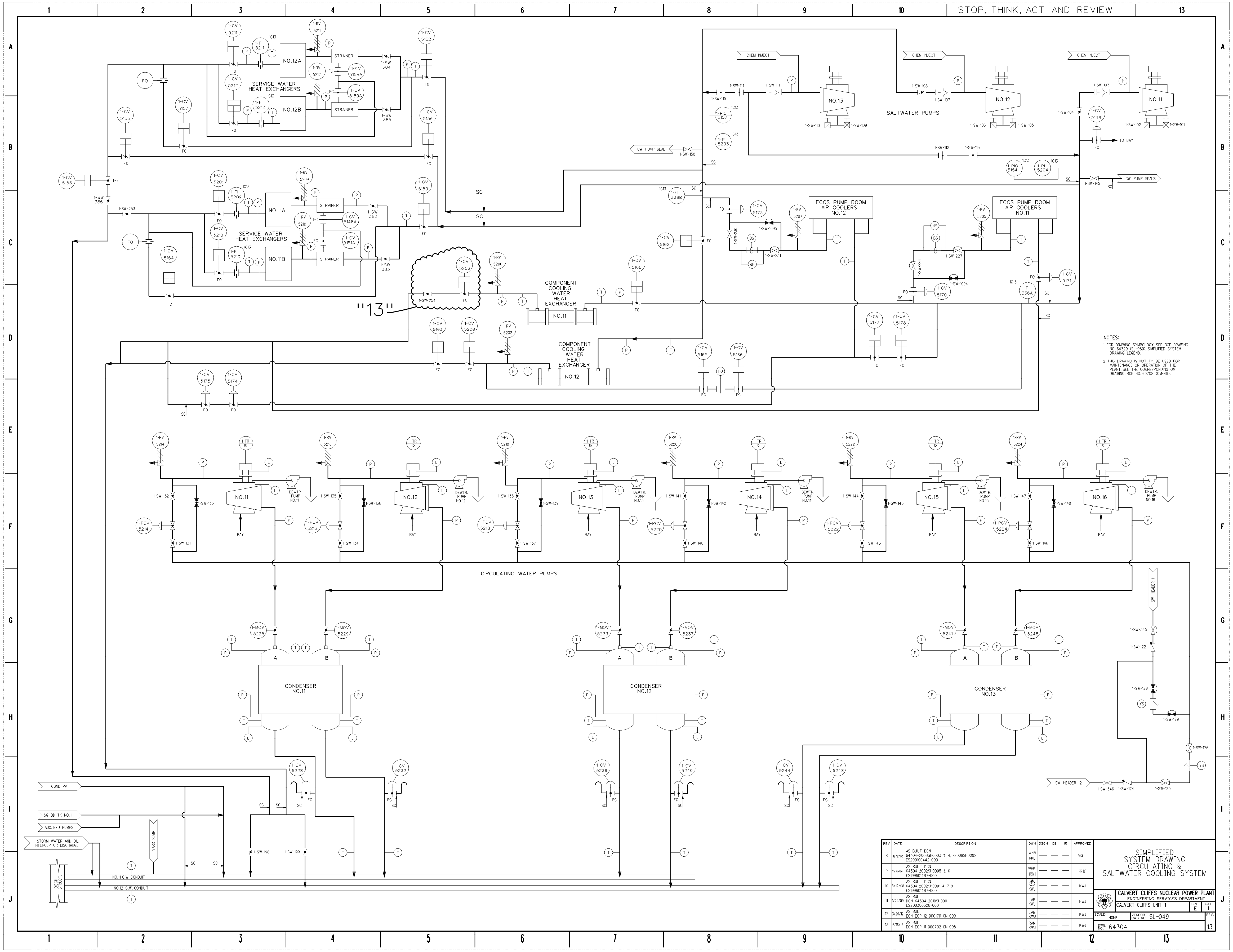


FIGURE 9-9 SERVICE WATER - UNIT 1

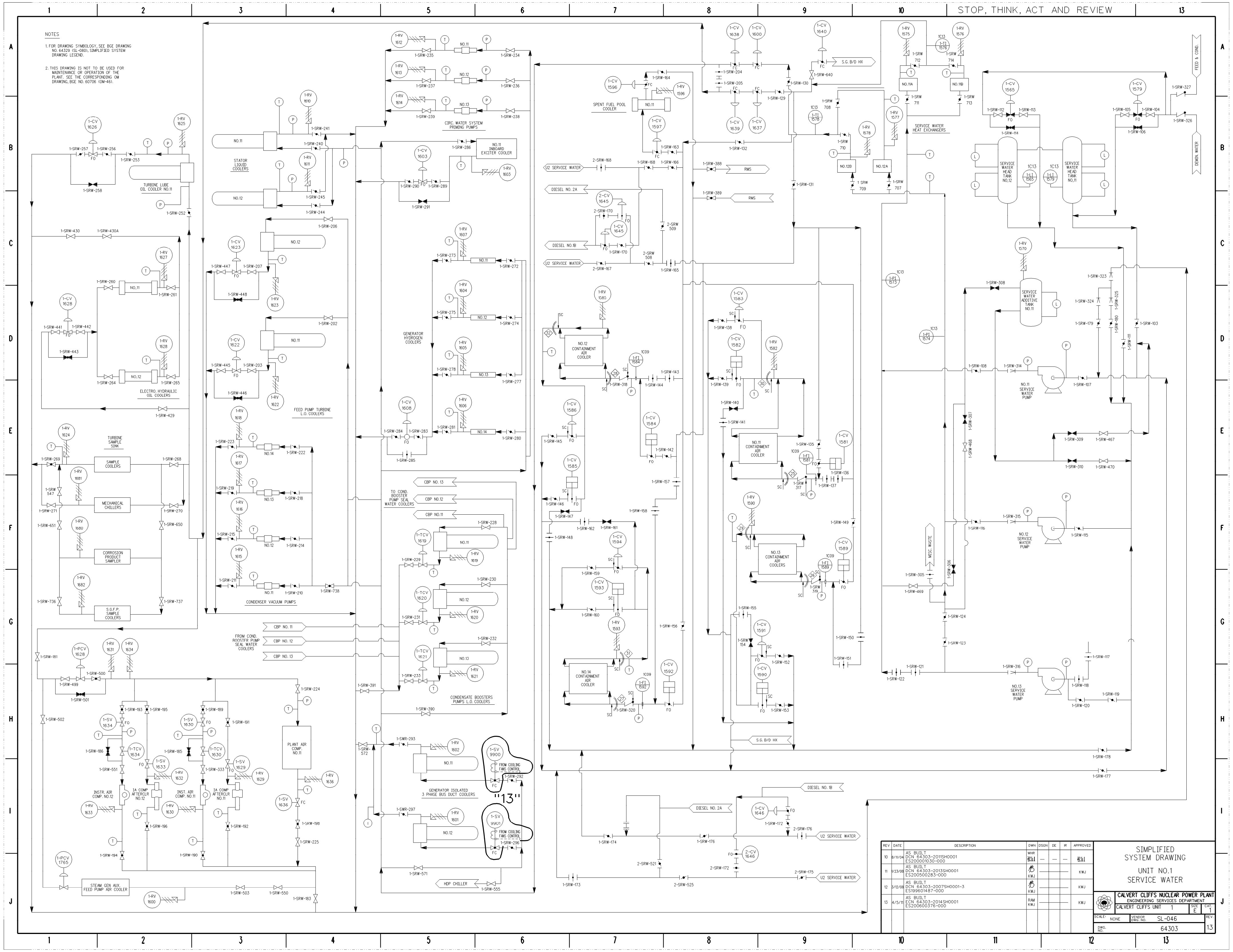
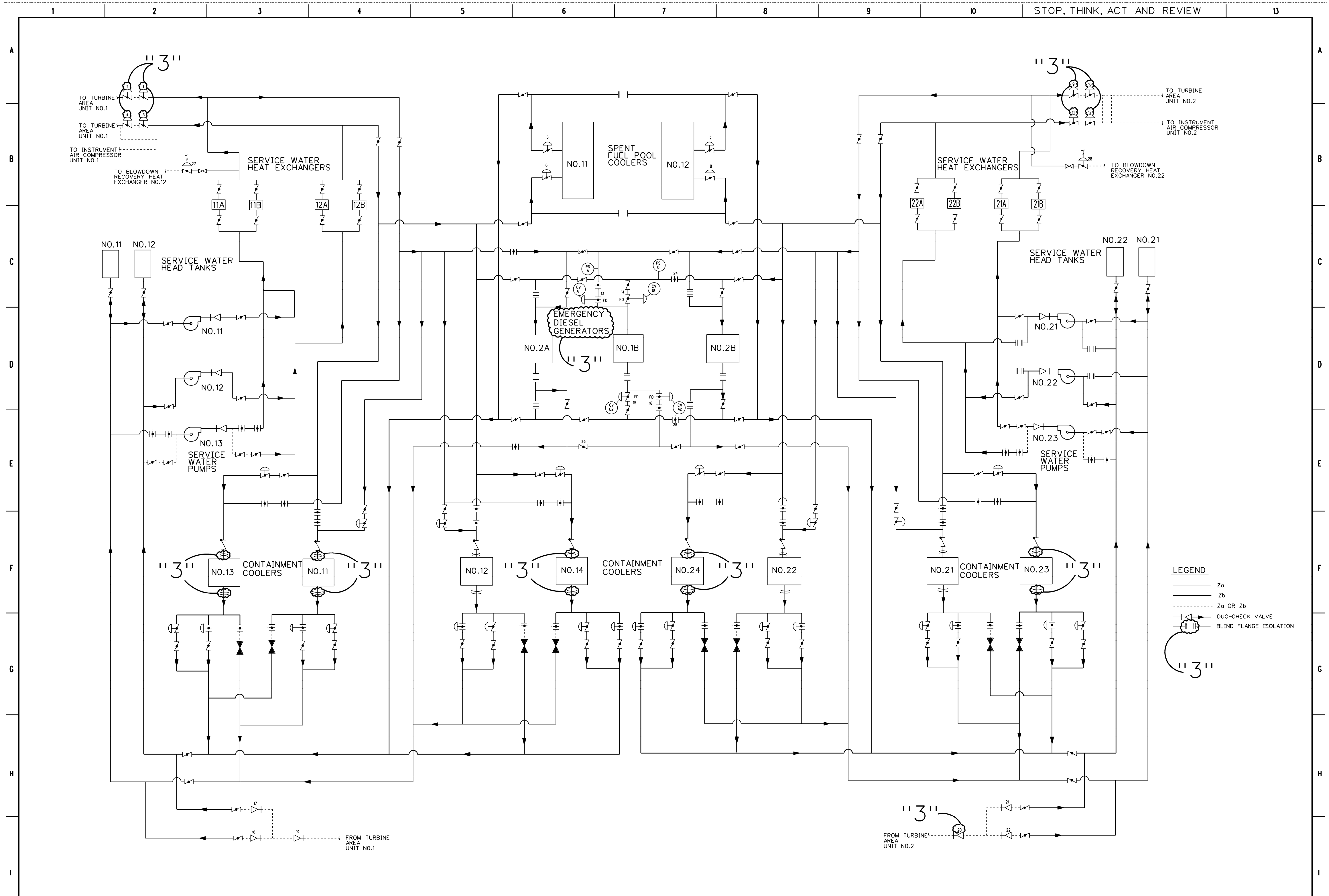


FIGURE 9-9B SERVICE WATER SYSTEM – CONTAINMENT AND AUXILIARY BUILDING UNITS 1 AND 2



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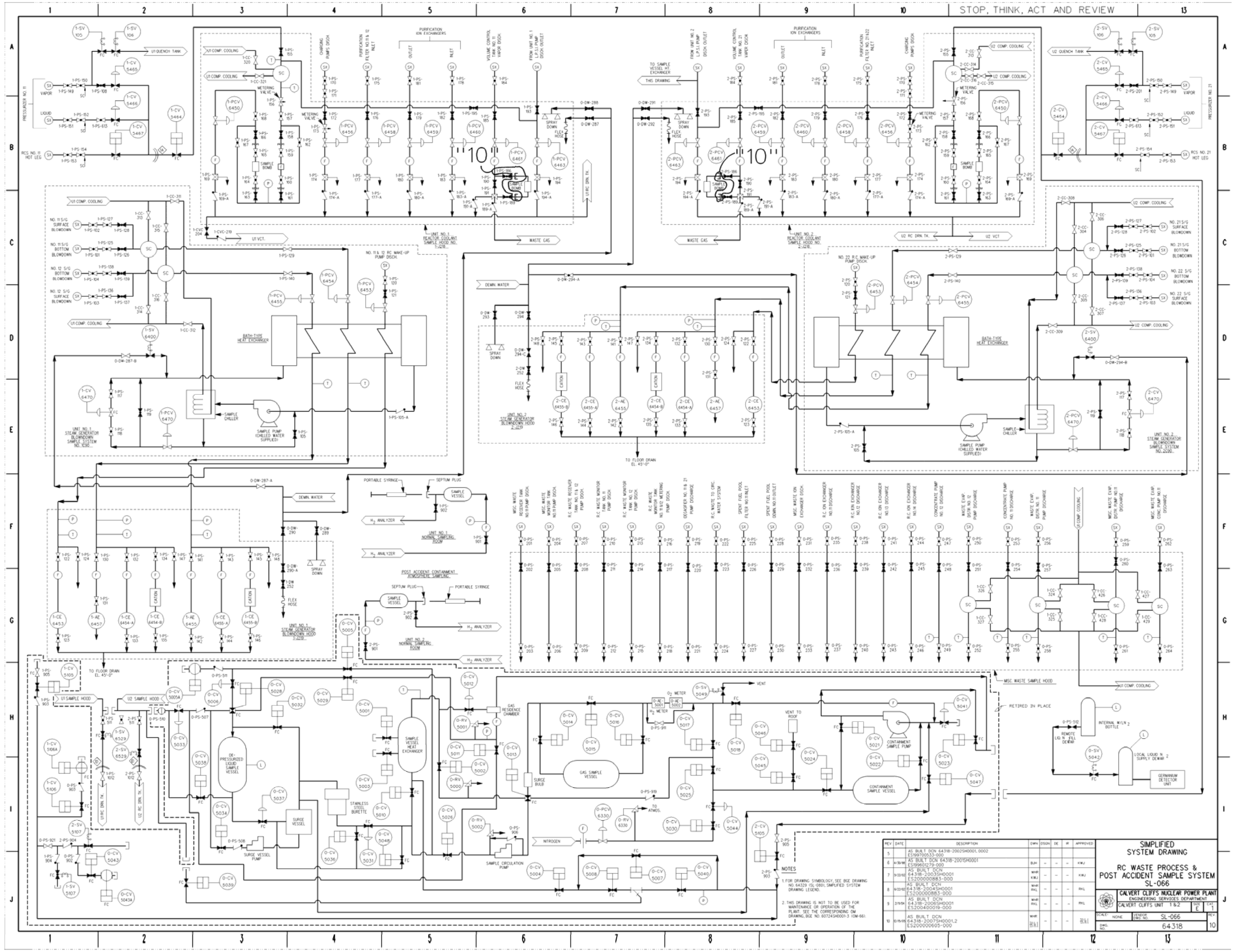
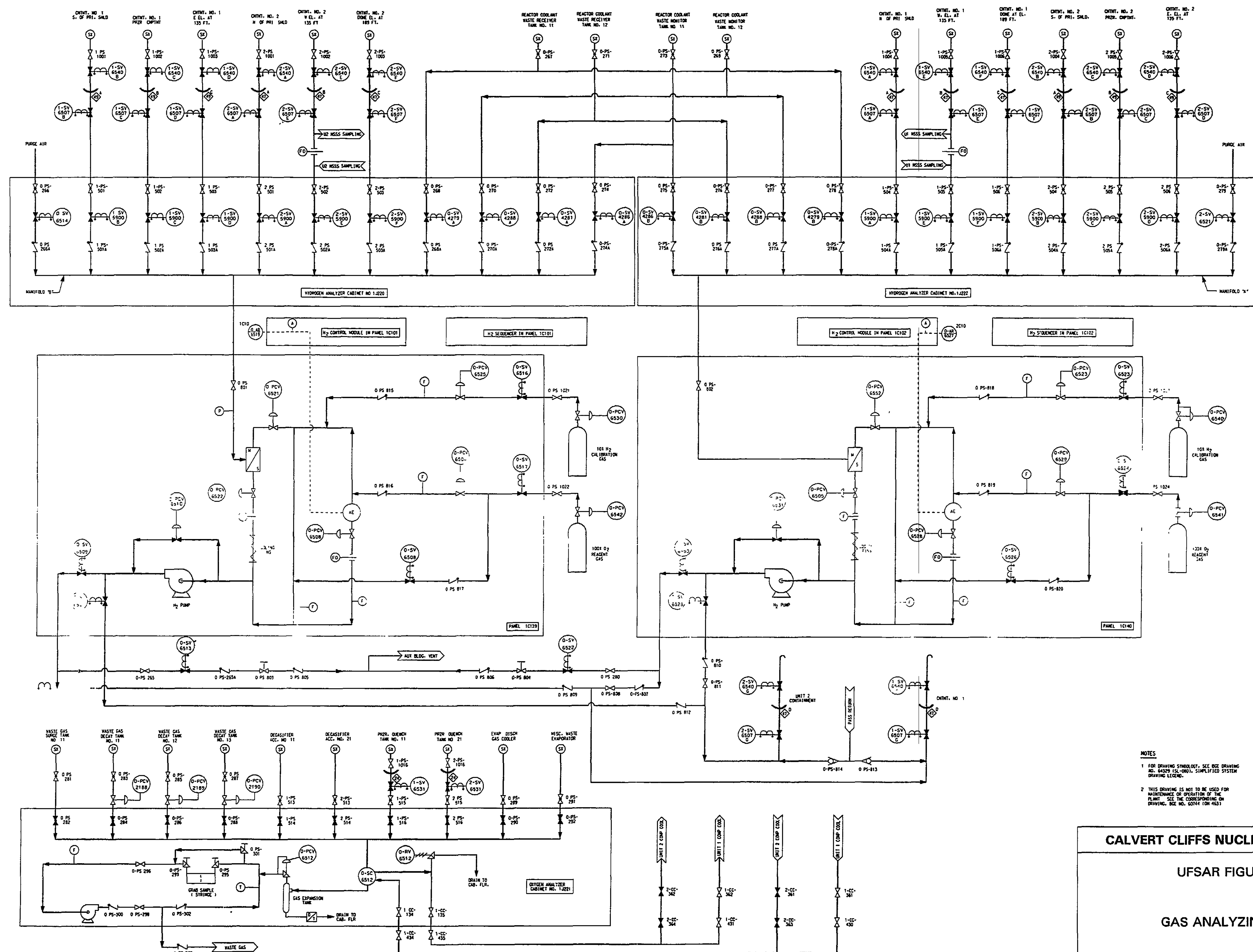
1	11/12/98	AS BUILT INFO ONLY ES199801456-000	MVR SF	—	—	—	JM
2	11/28/07	AS BUILT DCN SK-M-0329-0001 ES200000998-000	TJU KWJ	—	—	—	KWJ
3	6/4/14	AS BUILT INFO ONLY ADC-14-000524	DHE RAM	—	—	—	KWJ

SERVICE WATER SYSTEM
CONTAINMENT &
AUXILIARY BUILDING

CALVERT CLIFFS NUCLEAR POWER PLANT
ENGINEERING SERVICES DEPARTMENT
CALVERT CLIFFS UNIT 1&2 SIZE E CAT. 1
SCALE: NONE VENDOR DWG. NO. UFSAR FIG. 9-9B REV. 3
DWG. NO. SK-M-0329

FIGURE 9-10 REACTOR COOLANT/WASTE PROCESSING AND POST ACCIDENT SAMPLING SYSTEMS





CALVERT CLIFFS NUCLEAR POWER PLANT

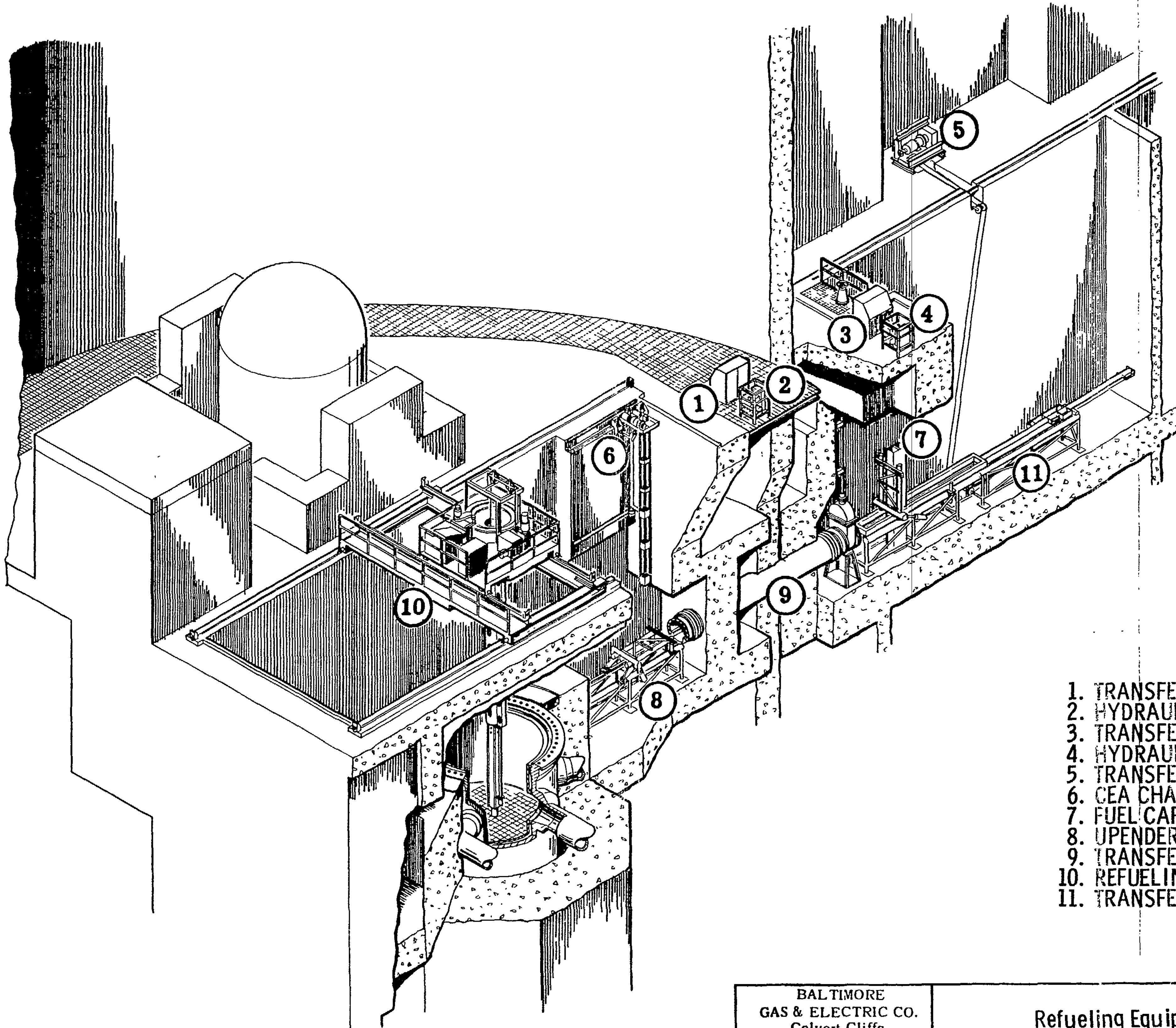
UFSAR FIGURE 9-11

GAS ANALYZING SYSTEM

BGE DRAWING 64-326, REV 1

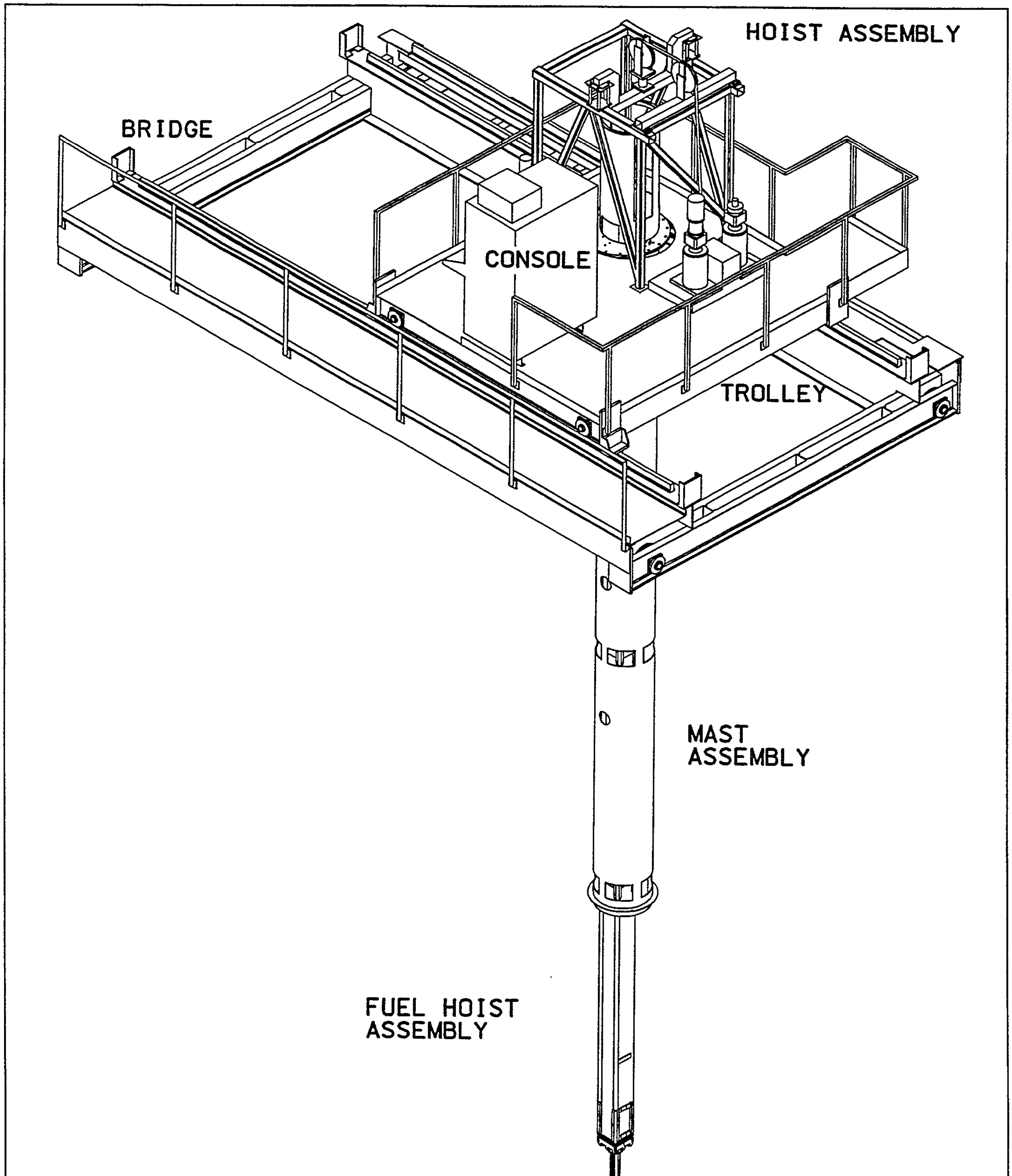
Revision 21

9
22



1. TRANSFER MACHINE CONSOLE
2. HYDRAULIC POWER PACKAGE
3. TRANSFER MACHINE CONSOLE
4. HYDRAULIC POWER PACKAGE
5. TRANSFER MACHINE WINCH ASSY
6. CEA CHANGE MACHINE
7. FUEL CARRIAGE
8. UPENDER
9. TRANSFER TUBE
10. REFUELING MACHINE
11. TRANSFER MACHINE

BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Refueling Equipment Arrangement	Figure 9 - 12



Calvert Cliffs Nuclear Power Plant	REFUELING MACHINE	Figure 9-13 Rev. 27

FIGURE 9-14 FUEL TRANSFER TUBE – UNITS 1 OR 2

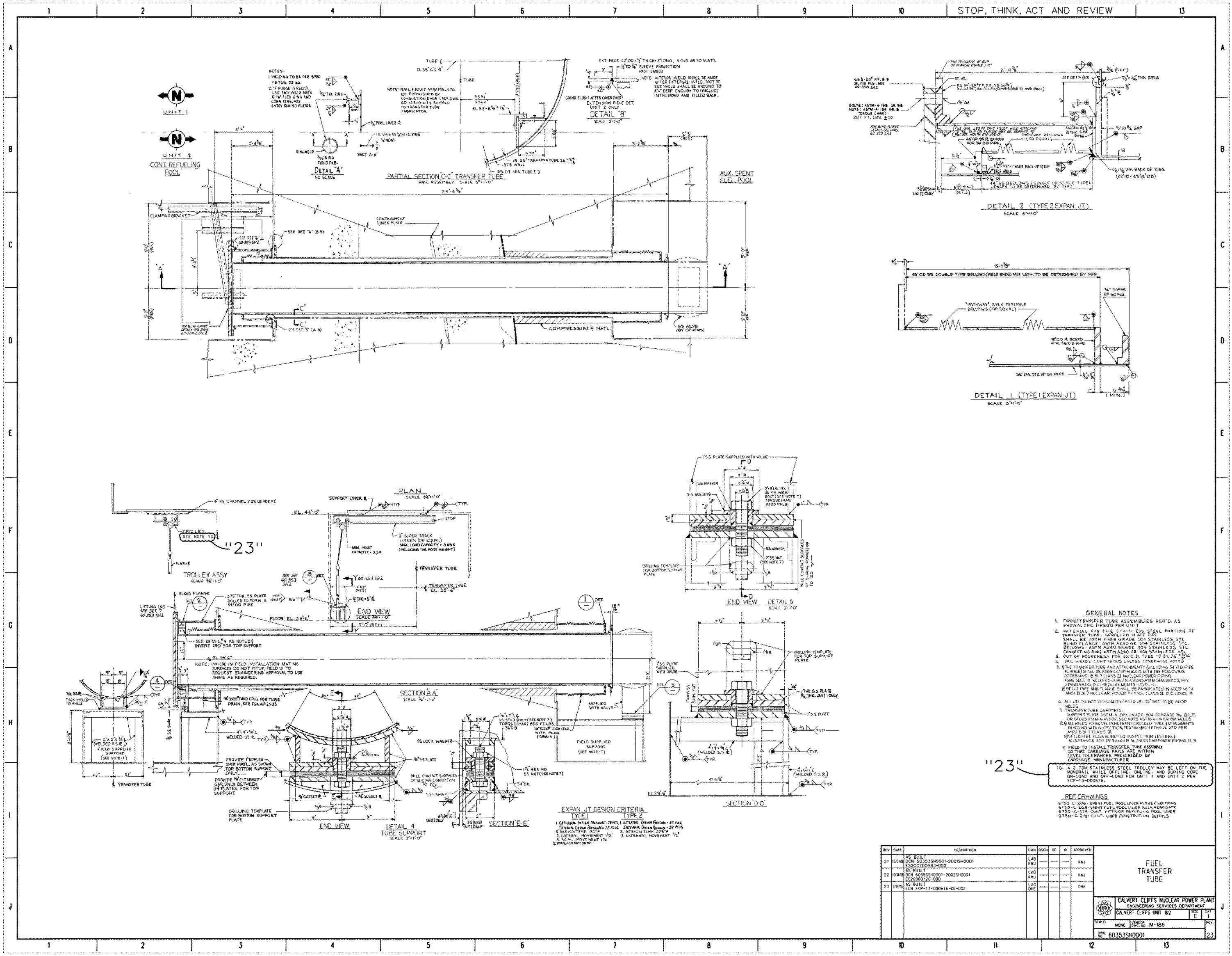
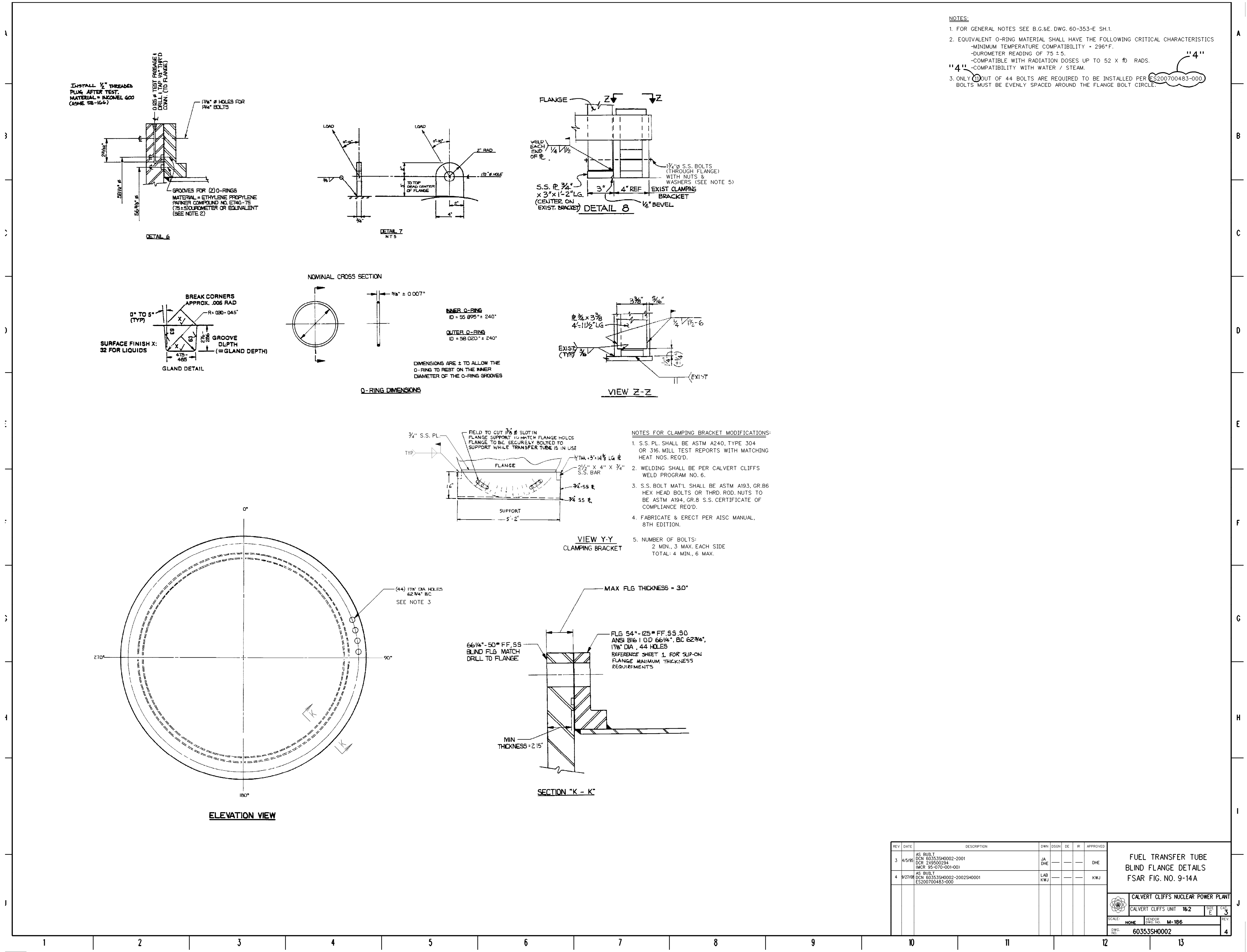
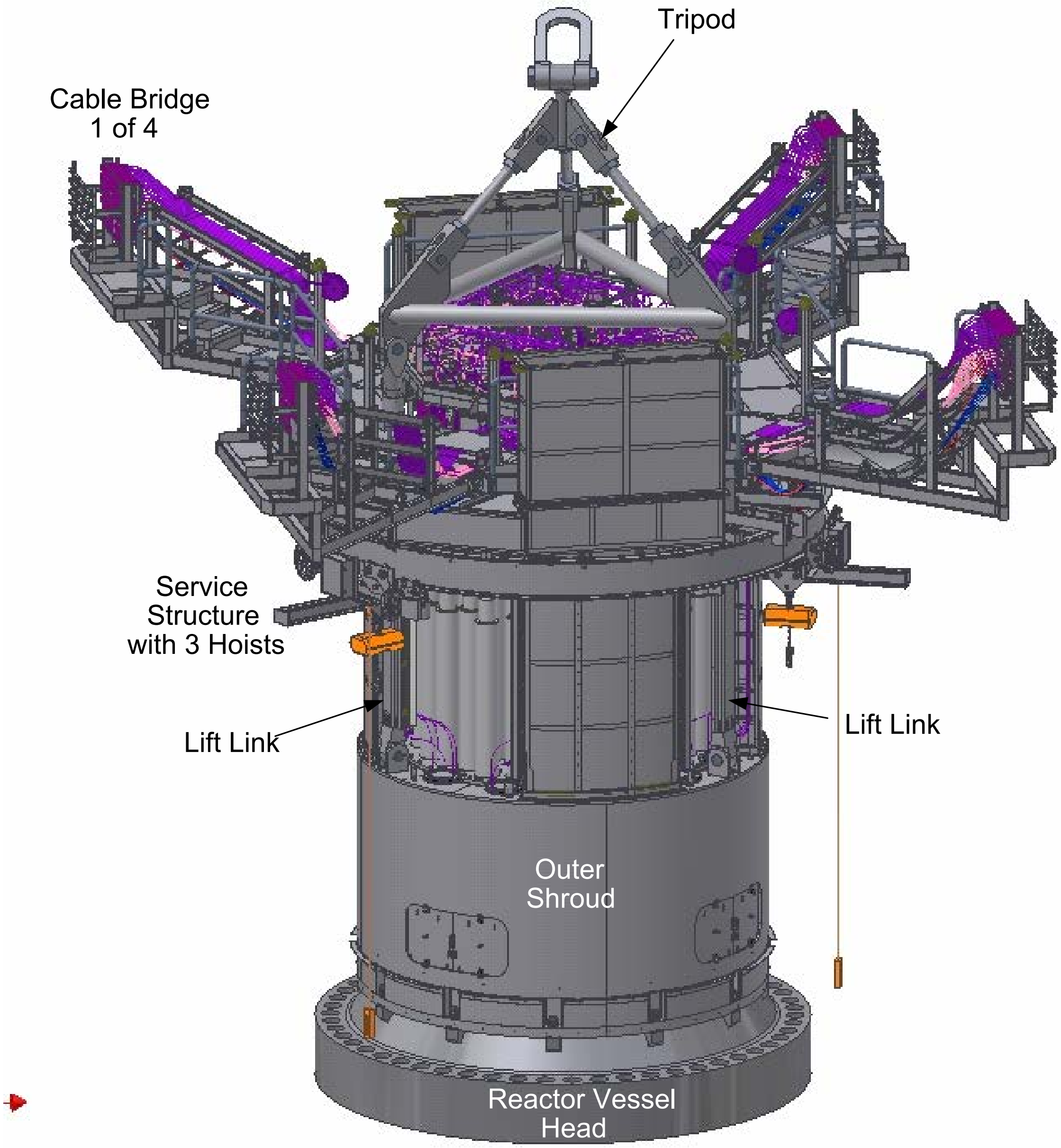
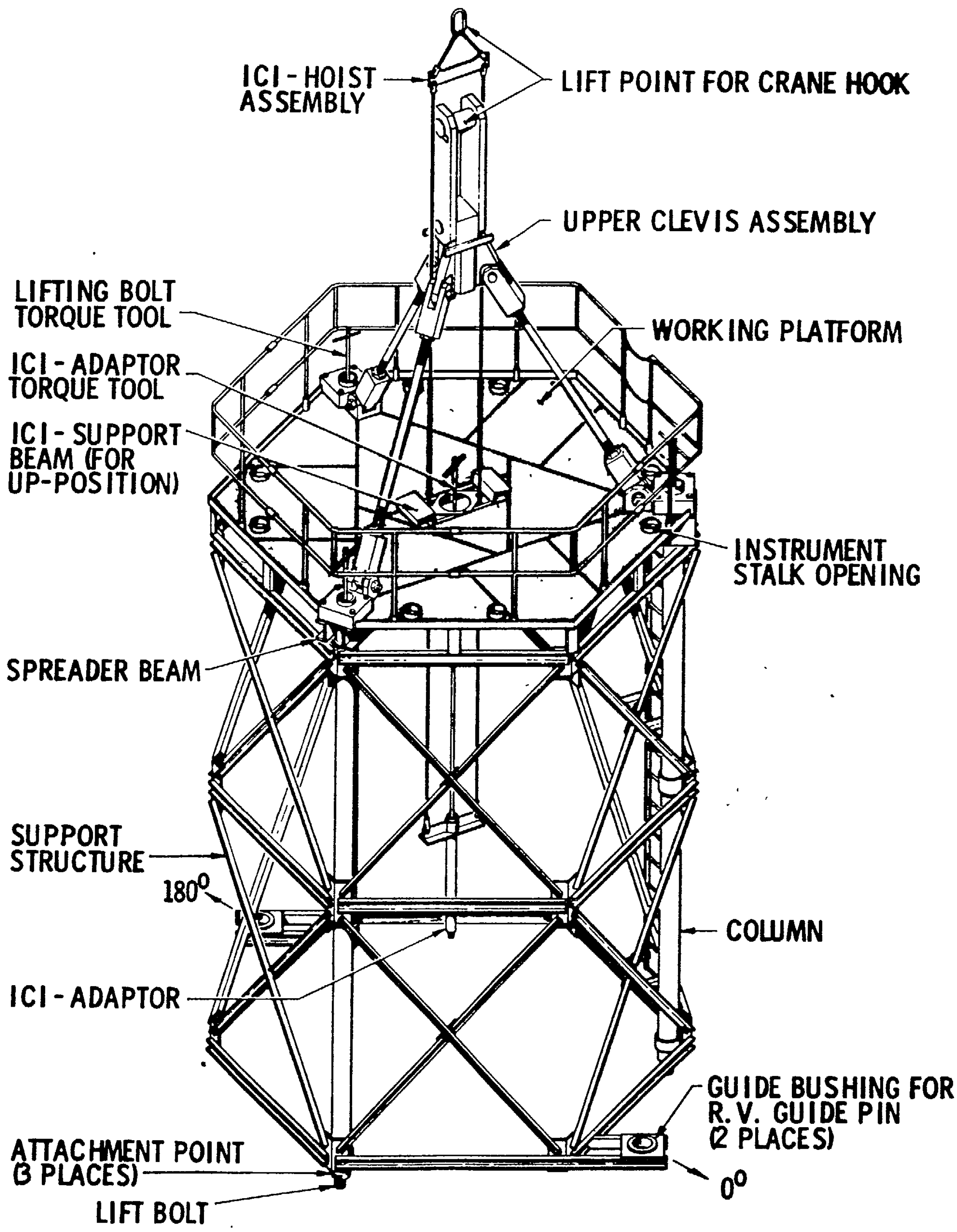


FIGURE 9-14A FUEL TRANSFER TUBE BLIND FLANGE DETAILS

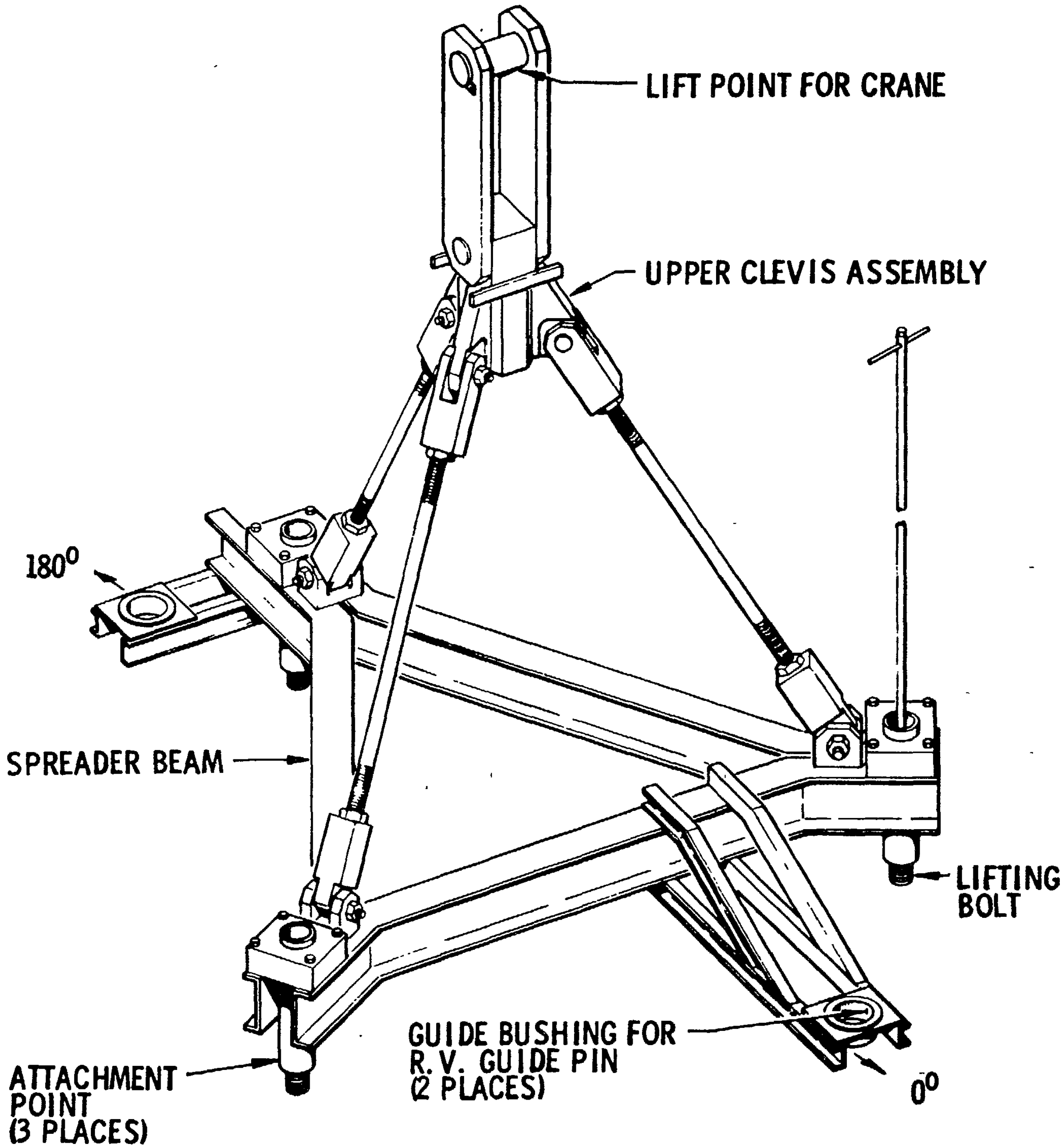




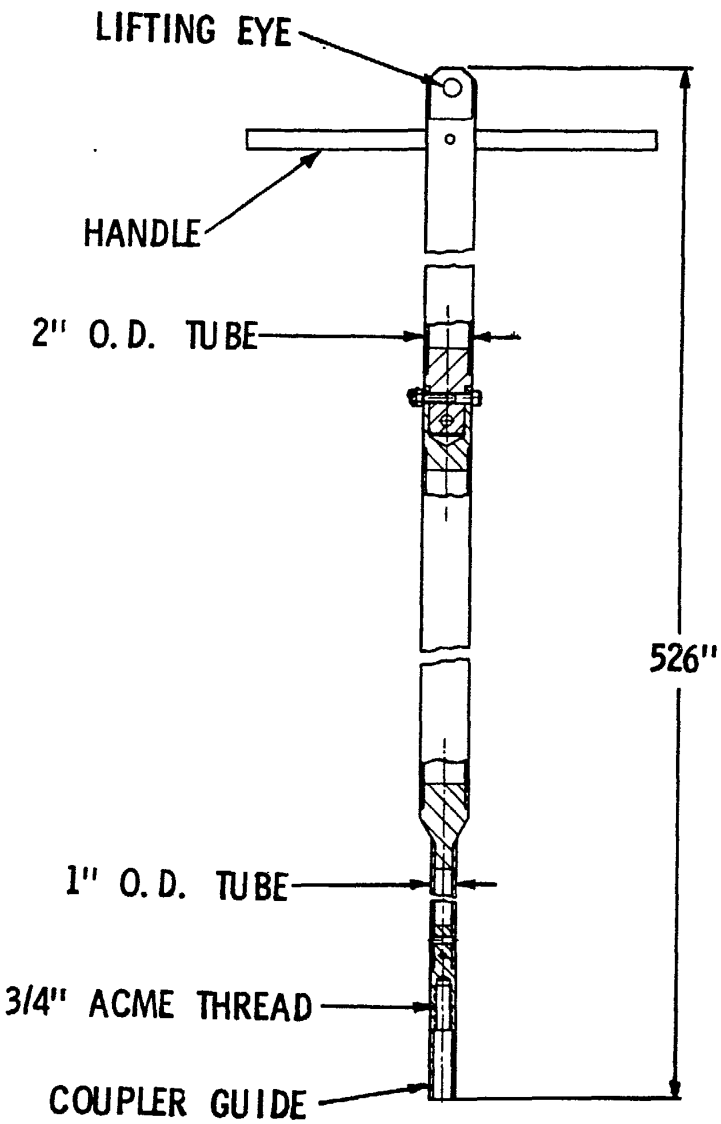
Calvert Cliffs Nuclear Power Plant	REPLACEMENT REACTOR VESSEL CLOSURE HEAD LIFTING RIG	Figure 9-16 Revision 39



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	UPPER GUIDE STRUCTURE LIFTING RIG	Figure 9-17



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	CORE SUPPORT BARREL LIFTING RIG	Figure 9-18



BALTIMORE GAS & ELECTRIC CO. Calvert Cliffs Nuclear Power Plant	Surveillance Capsule Retrieval Tool	Figure 9-19

FIGURE 9-20A CONTAINMENT AND PENETRATION ROOM VENTILATION

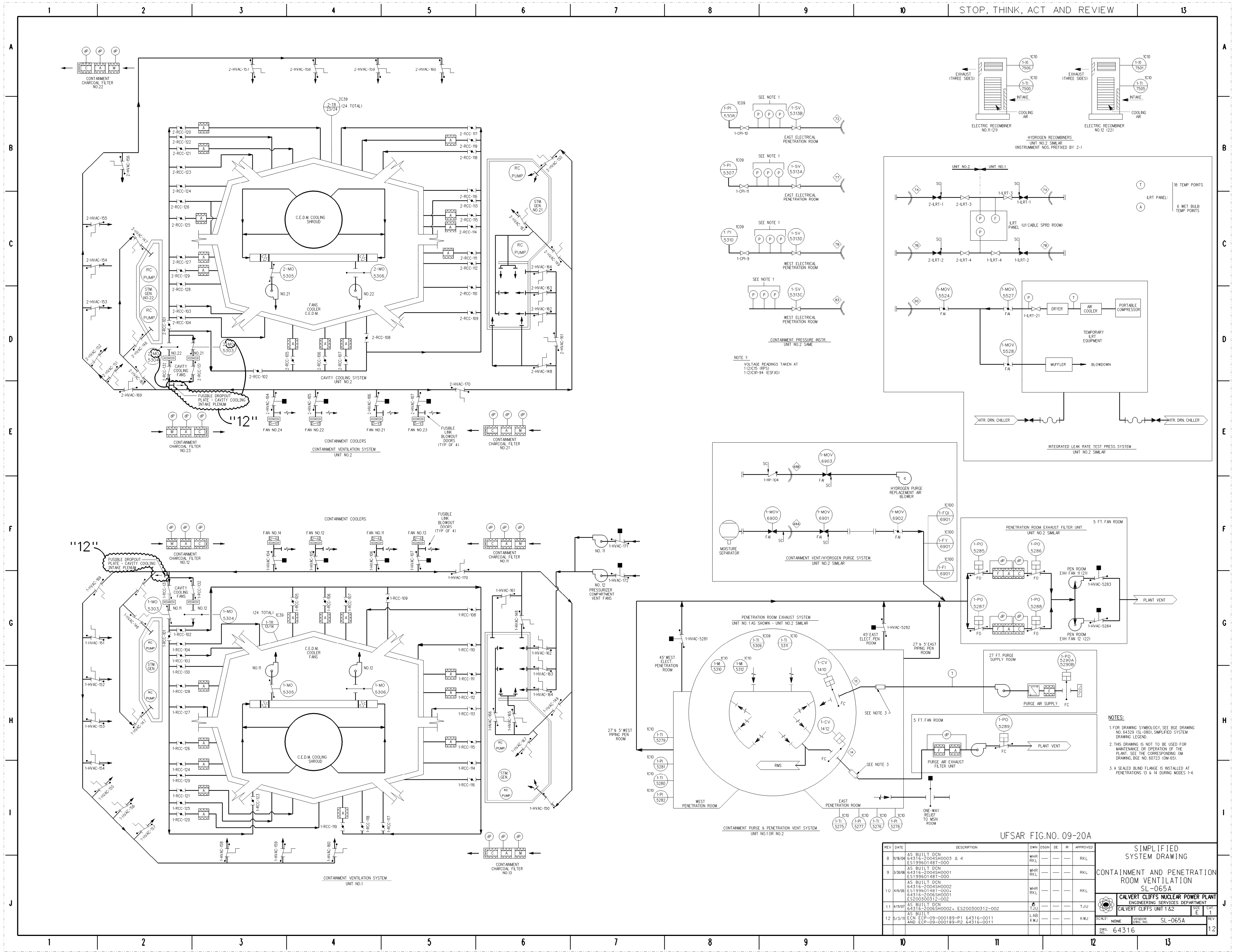
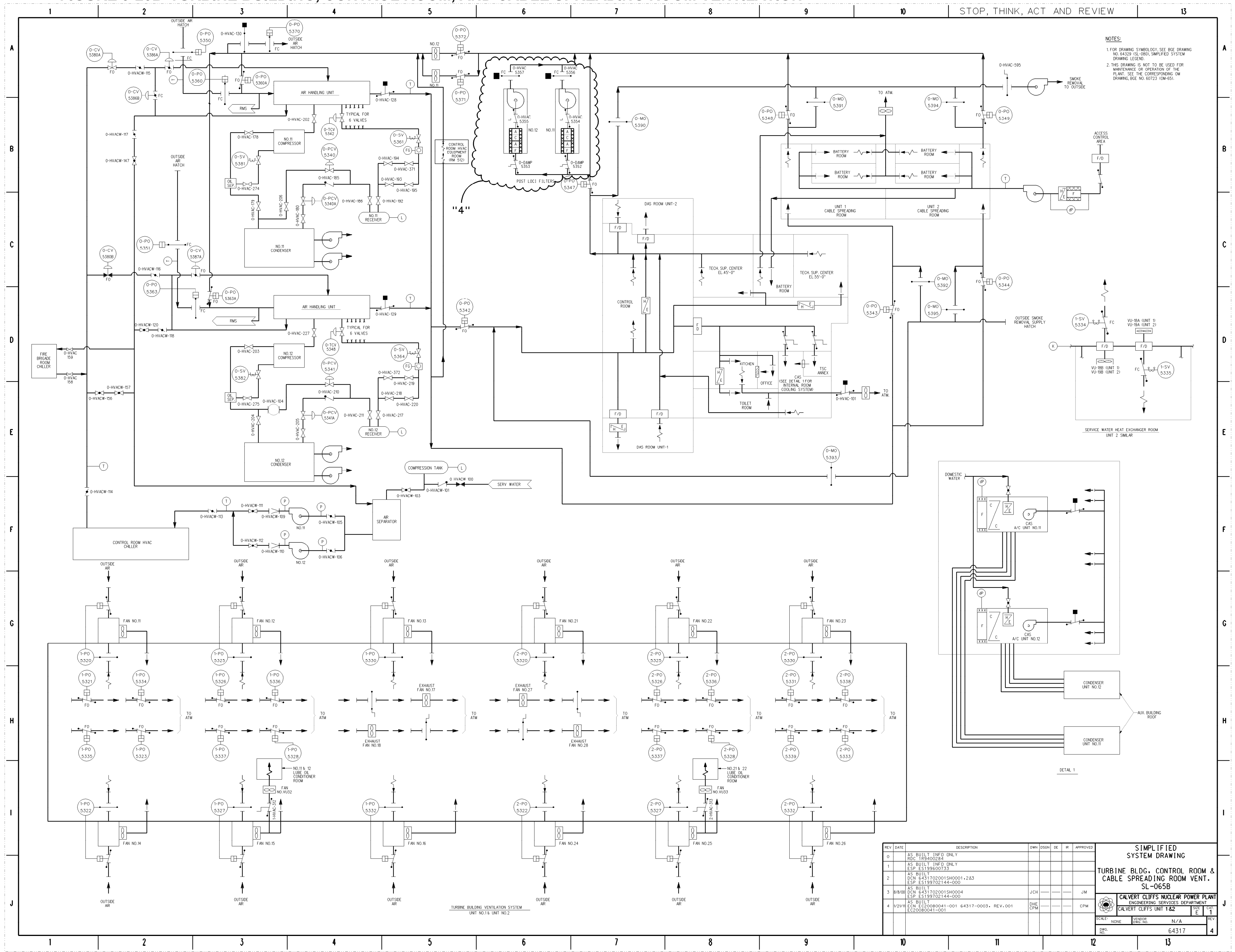


FIGURE 9-20B TURBINE BUILDING, CONTROL ROOM, AND CABLE SPREADING ROOM VENTILATION



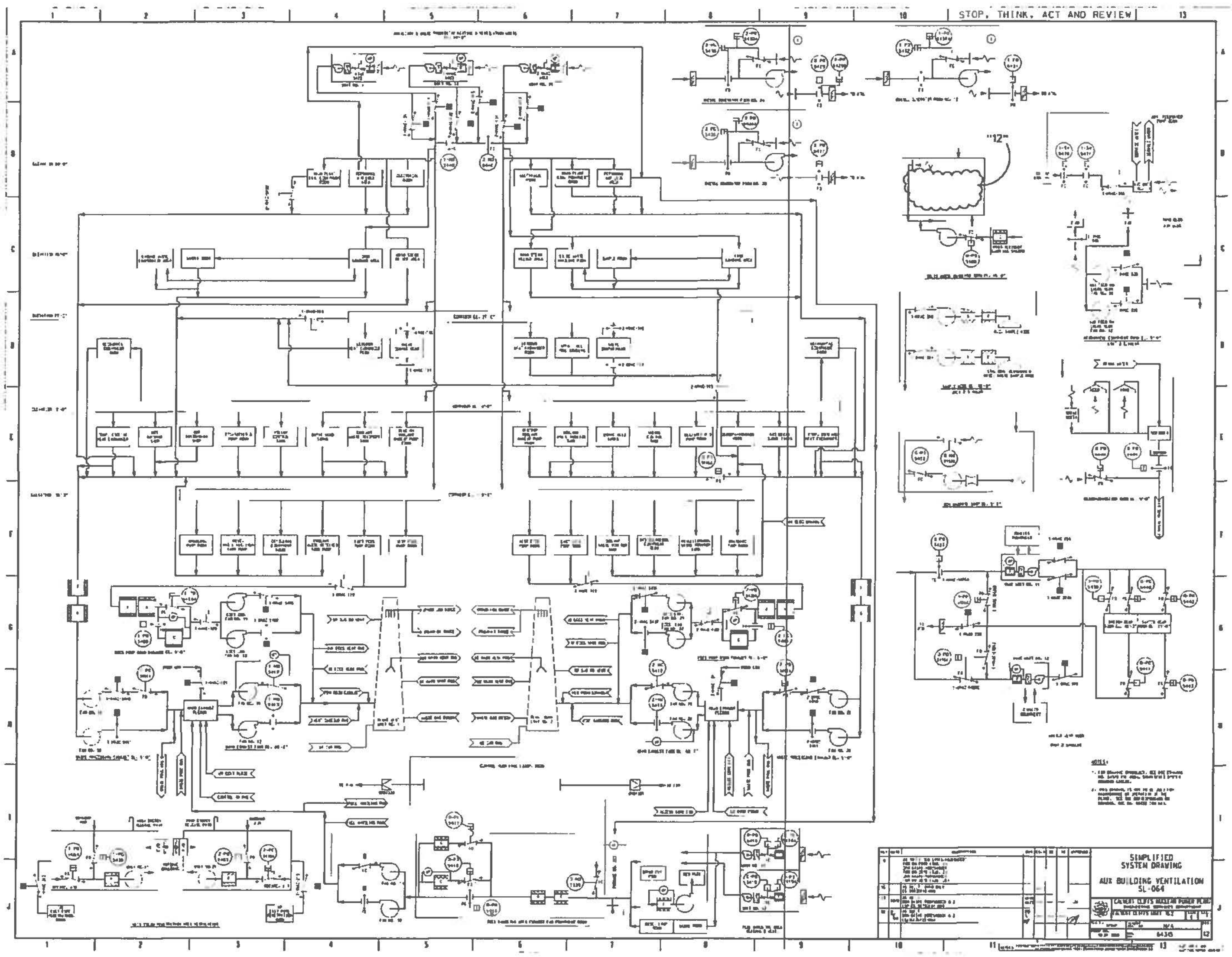

STOP, THINK, ACT AND REVIEW
"12"
SIMPLIFIED
SYSTEM DRAWING
AUX BUILDING VENTILATION
SL-064
CALVERT CLIFFS NUCLEAR POWER PLANT
N/A
64345

FIGURE 9-22 PLANT FIRE PROTECTION

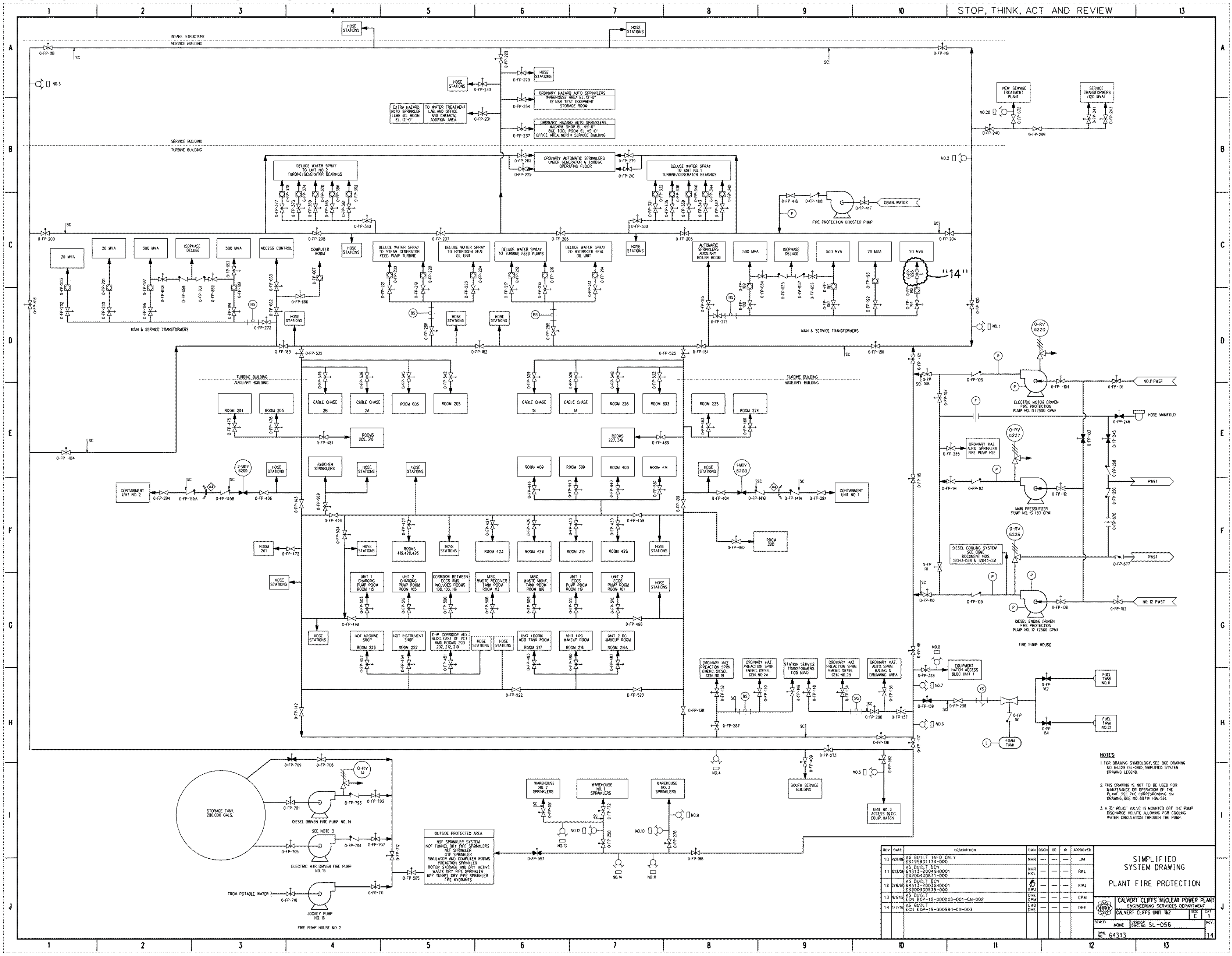


FIGURE 9-23 COMPRESSED AIR – UNIT 1

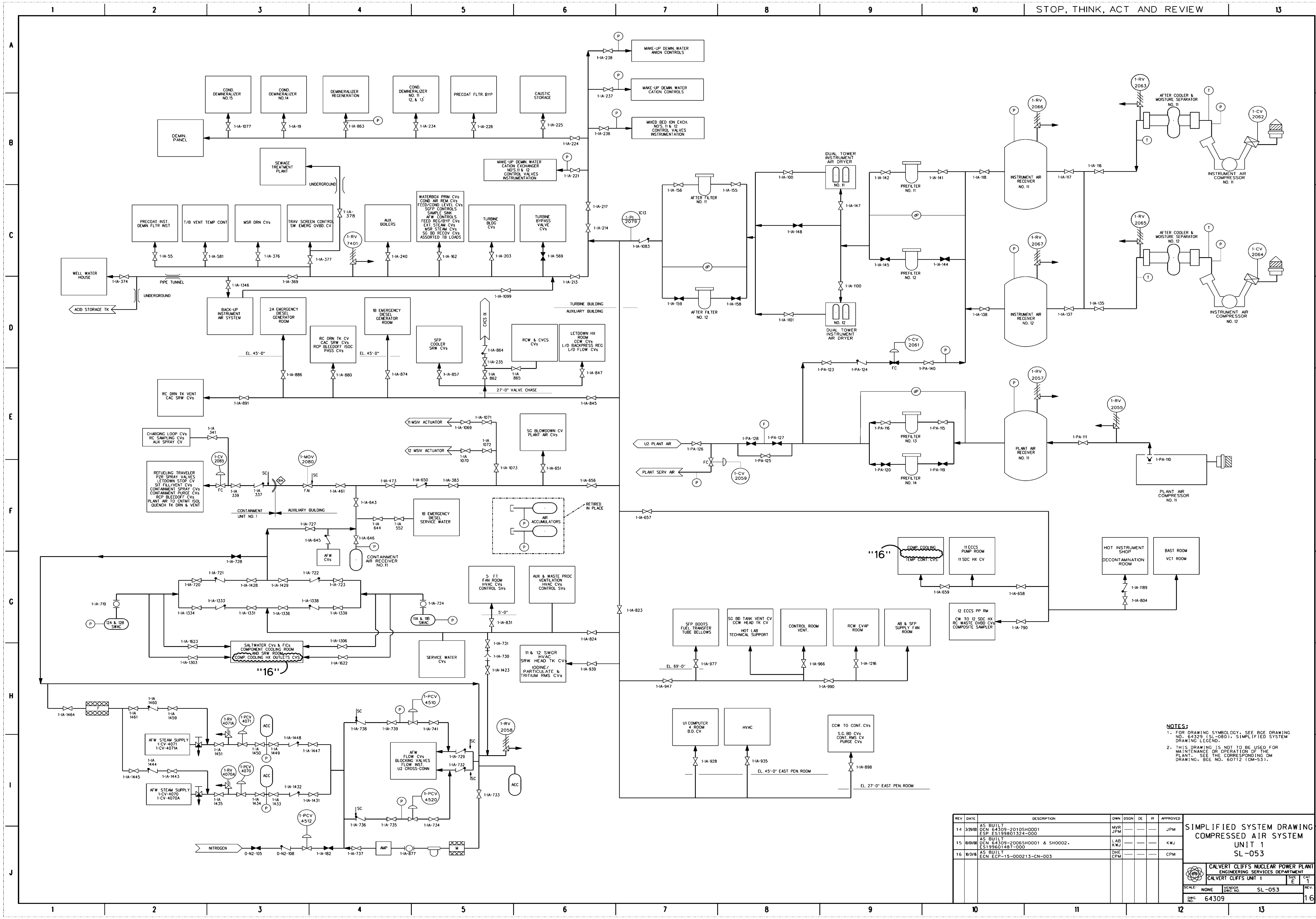


FIGURE 9-24 CHEMICAL AND VOLUME CONTROL SYSTEM – UNIT 2

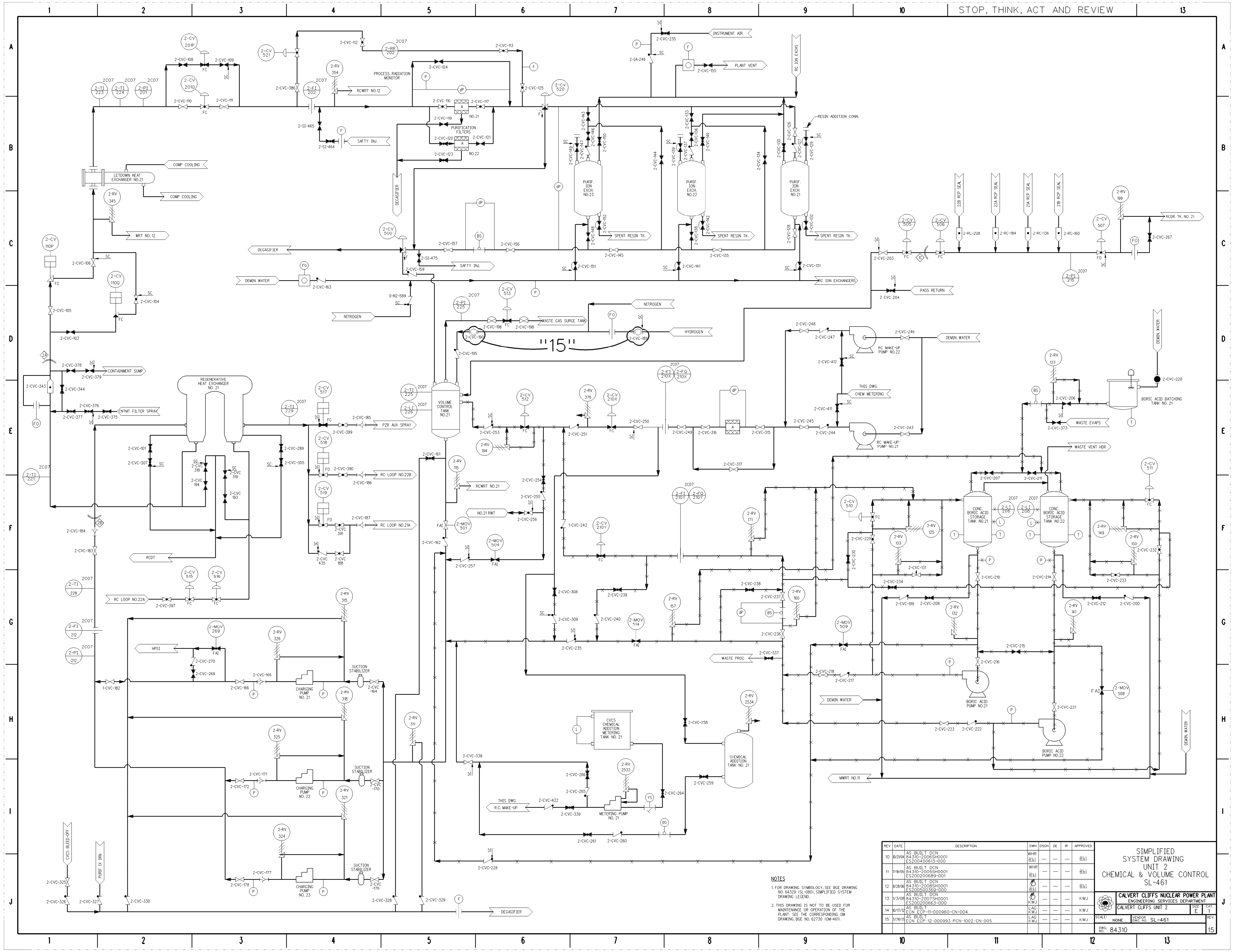


FIGURE 9-25 COMPONENT COOLING WATER – UNIT 2

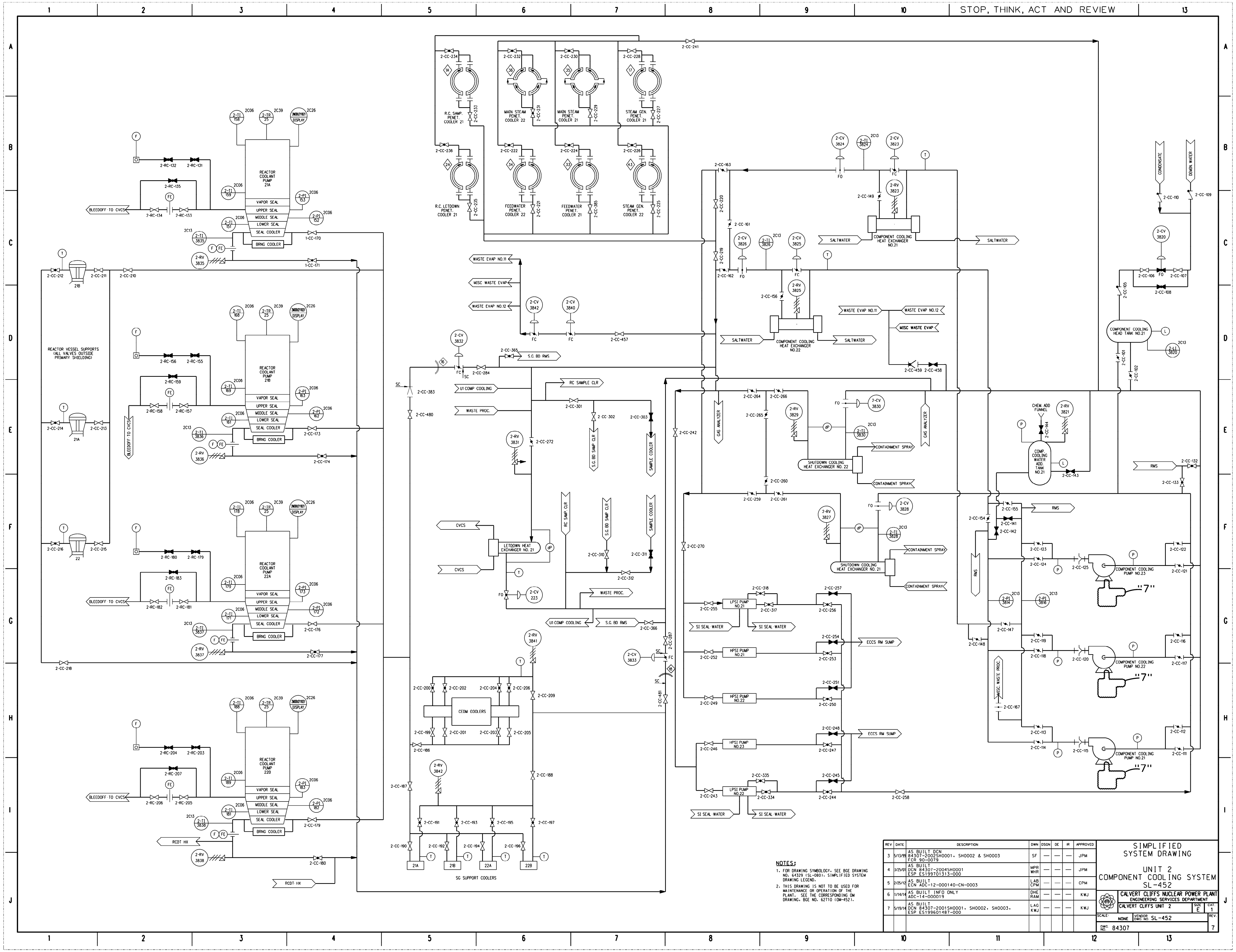
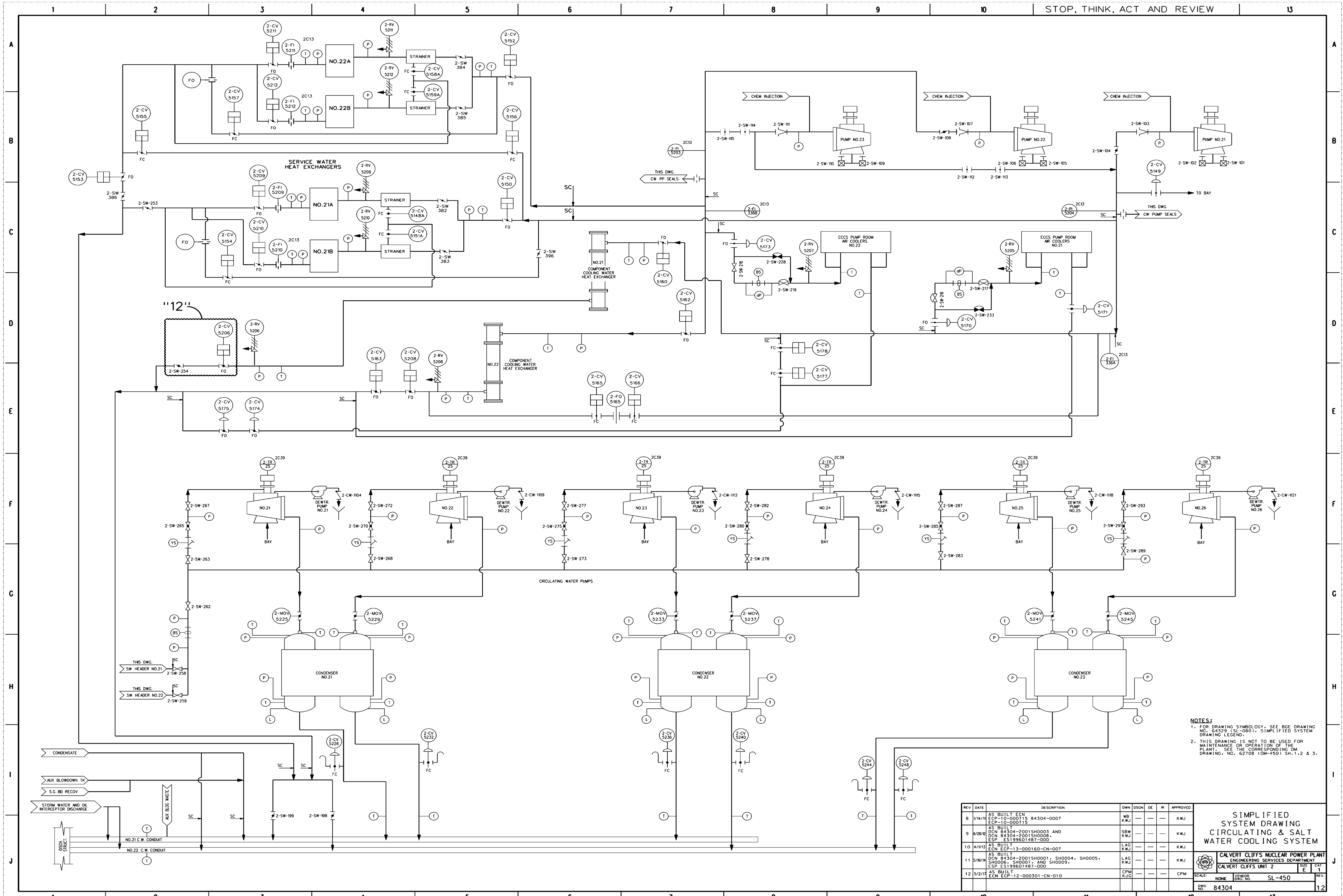


FIGURE 9-26 CIRCULATING AND SALTWATER – UNIT 2



REV	DATE	DESCRIPTION	DWN	DSGN	DE	IR	APPROVED
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9	6/28/12	AS BUILT DCN 84304-2001SH0003 AND DCN 84304-2001SH0008, ESP ES199601487-000	SBW KWJ	—	—	—	KWJ
10	4/1/13	AS BUILT ECN ECP-13-000160-CN-007	LAG KWJ	—	—	—	KWJ
11	5/19/14	AS BUILT DCN 84304-2001SH0001, SH0004, SH0005, SH0006, SH0007, AND SH0009, ESP ES199601487-000	LAG KWJ	—	—	—	KWJ
12	5/2/17	AS BUILT ECN ECP-12-000301-CN-010	CPM KJG	—	—	—	CPM

FIGURE 9-27 SERVICE WATER – UNIT 2

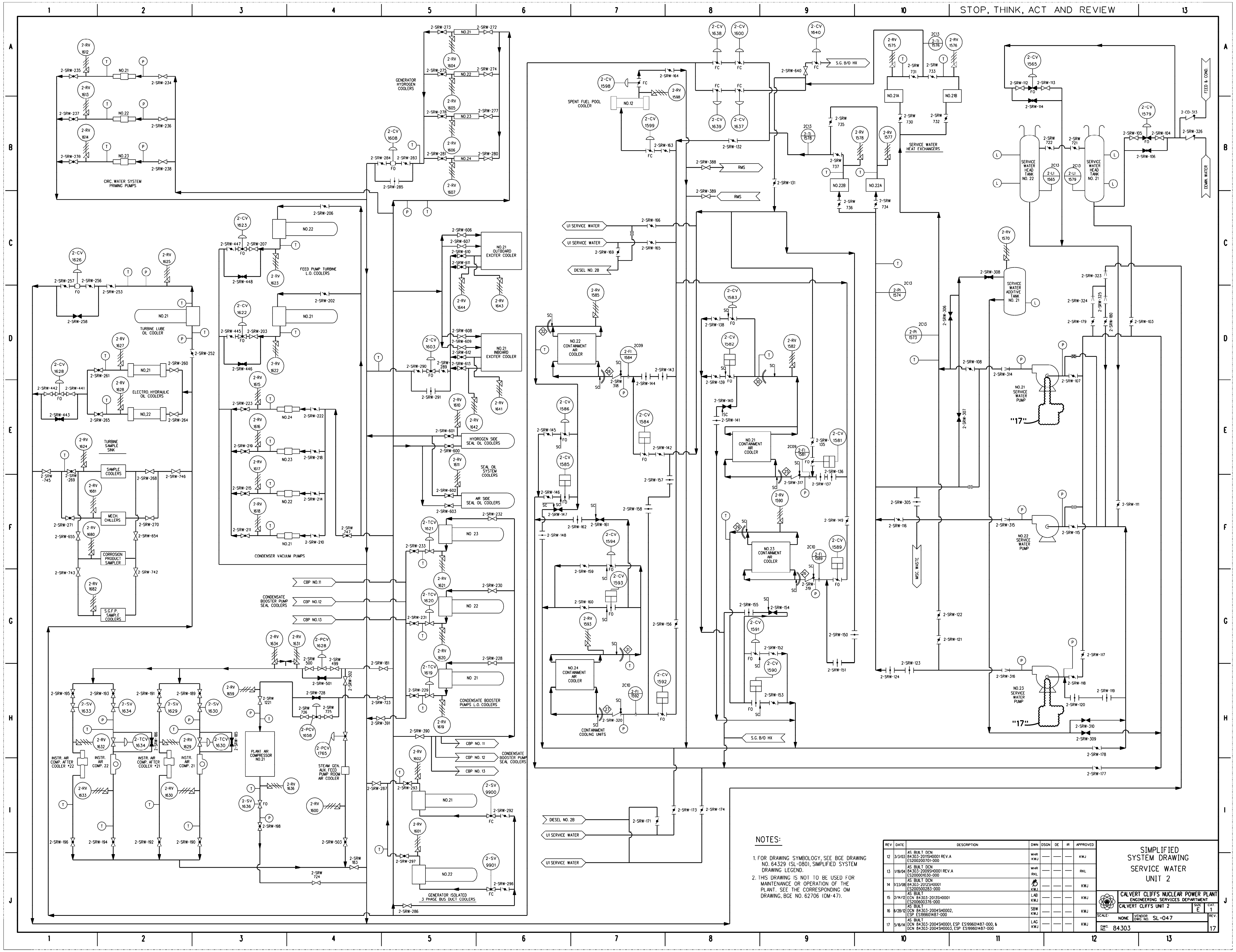

FIGURE 9-28 COMPRESSED AIR - UNIT 2

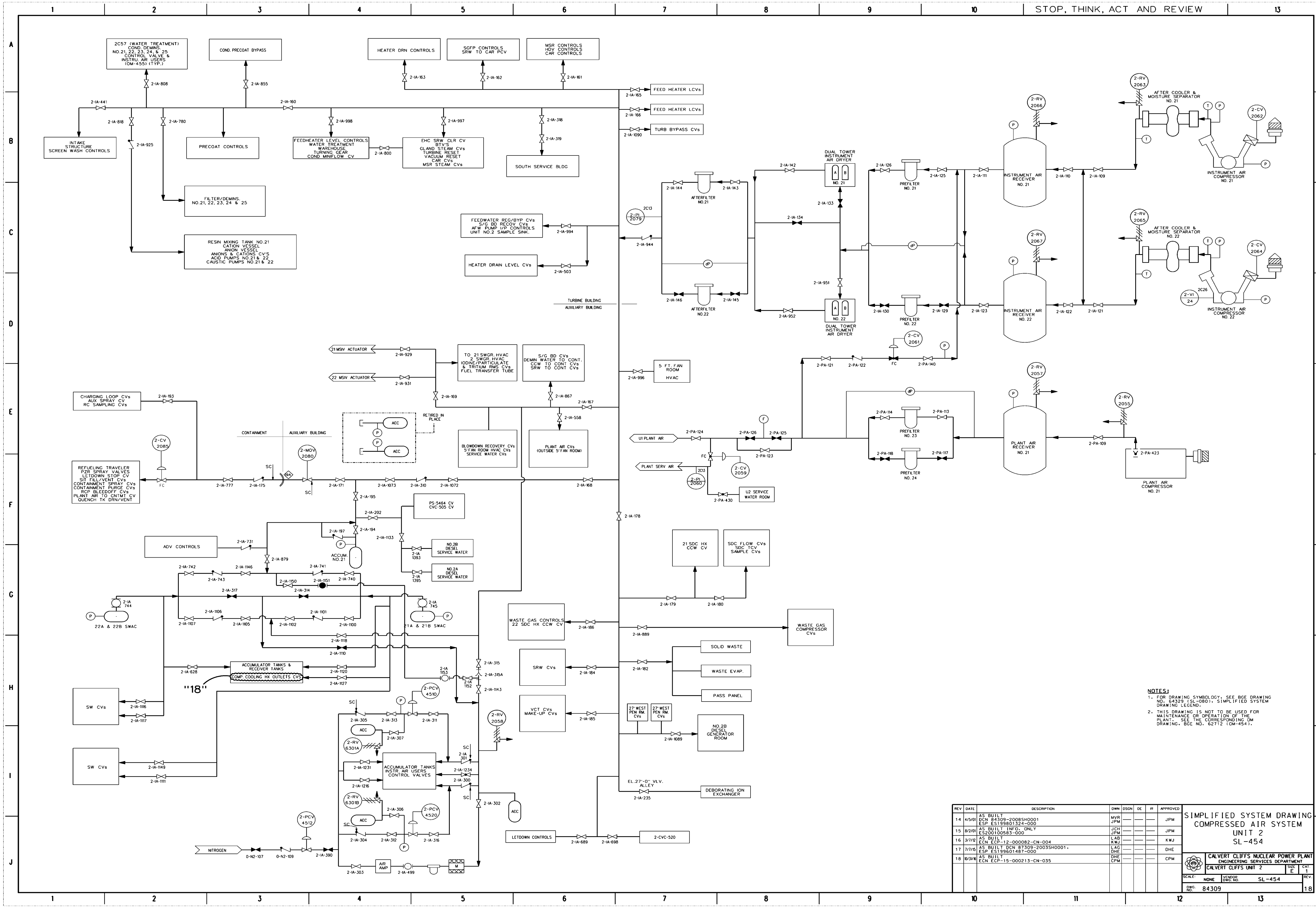

FIGURE 9-29 PLANT WATER AND AIR SERVICE SYSTEM

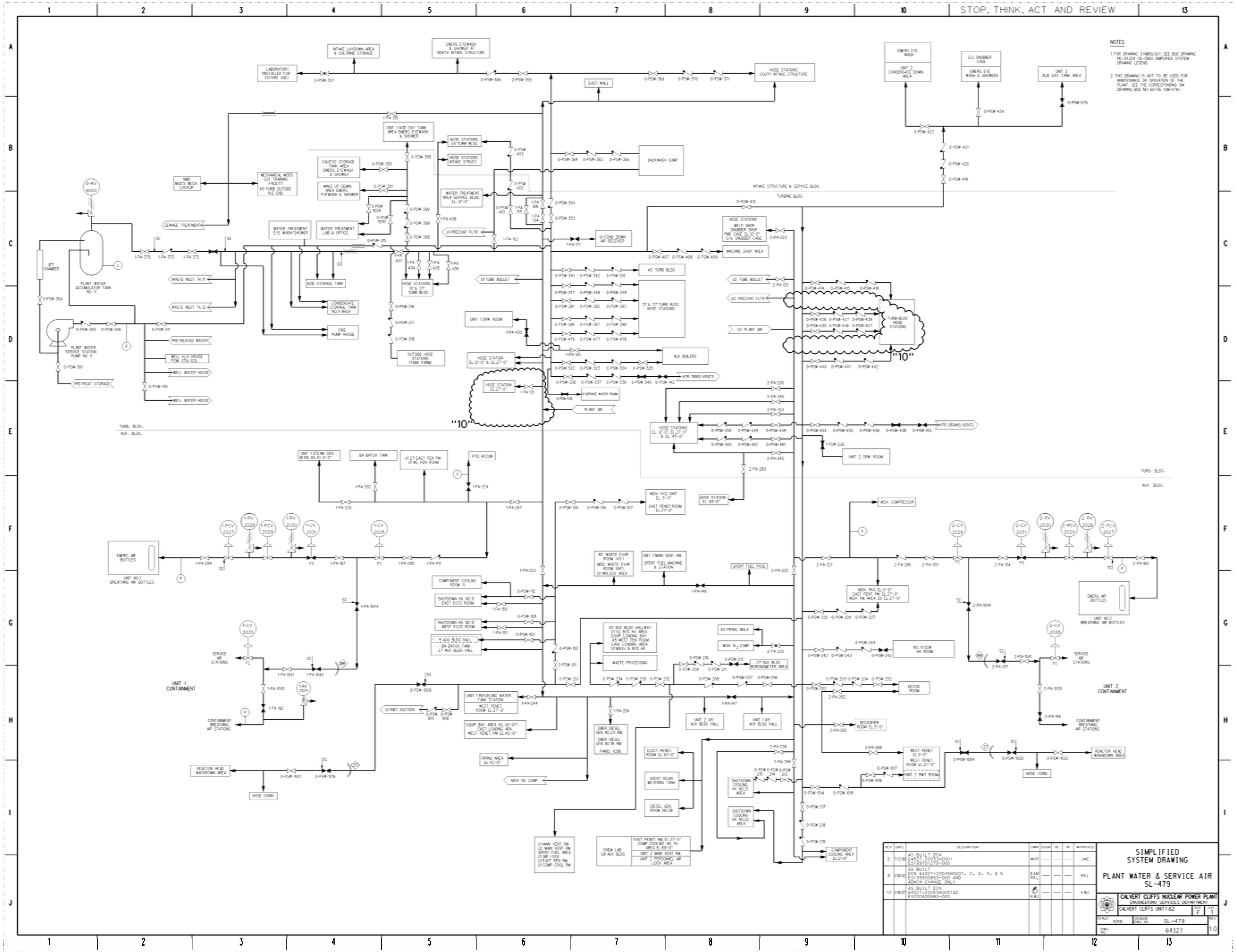

FIGURE 9-30 TURBINE SAMPLING SYSTEM